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Post-Severe Accident Environmental Conditions for Essential Instrumentation for Boiling Water Reactors

Dwight Clayton
Mike Poore

September 2015

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Electrical and Electronics Systems Research Division

POST-SEVERE ACCIDENT ENVIRONMENTAL CONDITIONS FOR ESSENTIAL INSTRUMENTATION FOR BOILING WATER REACTOR (BWR/4 MARK I CONTAINMENT DESIGNS

Dwight Clayton
Mike Poore

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ACRONYMS

ac	alternating current
BWR	boiling water reactor
CFR	Code of Federal Regulations
CV	control volume
dc	direct current
FSAR	Final Safety Analysis Report
IEEE	International Institute of Electrical and Electronics Engineers, Inc.
I&C	Instrumentation and Controls
LOCA	loss-of-coolant accident
LTSBO	long-term station blackout
NRC	US Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
PWR	pressurized water reactor
RCIC	reactor core isolation cooling system
RHR	residual heat removal
RPV	reactor pressure vessel
SOARCA	State-of-the-Art Reactor Consequence Assessment
STSBO	short-term station blackout
TEPCO	Tokyo Electric Power Company
TMI	Three Mile Island

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EXECUTIVE SUMMARY

The objective of this research is to estimate the environmental conditions that essential instrumentation must survive to remain functional following risk-dominant severe accidents at boiling water reactor (BWR)/4-Mark I reactors to improve severe accident management capabilities. This could lead to improvements in instrumentation survivability and performance under challenging operating conditions—a major reason for this research program applicable to BWR/4-Mark I and potentially to other BWR designs.

This report is the third of three related reports that examines the performance of BWR/4-Mark I reactor instrumentation used to monitor and manage plant performance during severe accidents. The first report [1] described the significant loss of monitoring and instrumentation and control (I&C) systems as a result of the Fukushima Daiichi accident, initially due to loss of power and then to the harsh severe accident environmental conditions. A second report [2] examined the key parameters that instrumentation needs to measure to help operators respond to severe accidents. This report examines the post-accident environmental conditions that could affect the performance of essential instrumentation systems that guide severe accident management involving a core melt and reactor vessel breach.

Loss of electric power and harsh, but as yet unquantified, environmental conditions were experienced at the Fukushima Daiichi units that affected instrumentation performance. Instrumentation failed, provided inaccurate readings, or indicated misleading or inconsistent trends, thus complicating operator responses and accident management.

The MELCOR severe accident code was used to examine postulated unmitigated short-term station blackout (STSBO) and long-term station blackout (LTSBO) severe accident scenarios for a BWR/4 with a Mark I containment developed as part of the State-of-the-Art Reactor Consequence Assessment project sponsored by the U.S. Nuclear Regulatory Commission [4, 5] to improve reactor accident analysis efforts. These analysis results were used as a basis for estimating environmental conditions in the reactor primary containment and reactor building during these scenarios. Instrumentation that provides essential measurement parameters and values necessary for informing accident management and mitigation efforts is subject to these environmental conditions.

This study found that instrumentation for about 20 reactor parameters deemed critical to informing operator responses and accident management activities would have exceeded qualification values for the STSBO and LTSBO severe accident scenarios. Instrumentation located in the drywell would typically have exceeded qualification values for pressure and temperature. Instrumentation located in the wetwell would have exceeded qualification values for pressure and radiation dose. Instrumentation located in the reactor building would typically have exceeded qualification values for temperature and radiation dose. Instrument qualification values would typically have been exceeded in the early hours and days of the accident scenarios during the period of core damage, hydrogen generation, core relocation from the reactor vessel to the drywell, and breach of containment. It is not presumed that an instrument fails if it exceeds its environmental qualifications. However, effective accident response depends on knowledge of plant parameters and conditions and is challenged when instrument performance is unreliable or inaccurate. At a minimum, it should be used with suspicion.

The conclusions of the study, however, are tempered by several issues, including (1) uncertainties associated with the MELCOR code to estimate equipment environmental conditions, and (2) unavailability of forensics analysis of instrumentation at the Fukushima Daiichi stations at this time and the related subject of how instrumentation currently in use performs in harsh conditions.

Accident responses, such as through emergency procedures or severe accident management guidelines, that rely on instrumentation for critical parameters should recognize and account for potential widespread instrument failure or inaccurate and/or misleading instrument indications during these severe accident scenarios.

Future work may address improvements in the MELCOR models to provide better certainty in estimates of environmental conditions, especially radiation effects. The improvements could also be informed based on results of Fukushima Daiichi forensics analyses. Better knowledge of potential instrument performance under harsh conditions could lead to better instruments, alternative measurement technologies, and improvements in accident management strategies and plans.

ABSTRACT

The objective of this research is to estimate the environmental conditions under which essential instrumentation must survive to remain functional following risk-dominant severe accidents at boiling water reactors (BWRs), specifically a BWR/4-Mark I. Essential instrumentation systems are those considered to be most important for accident management purposes in the event of a severe accident involving a core melt and reactor vessel breach. Better understanding of the severity and duration of harsh environmental operating conditions associated with severe accidents can inform decisions about instrumentation performance limitations, degradation mechanisms, and consequences. This understanding could lead to improvements in instrumentation survivability and performance under challenging operating conditions—a major reason for this research program—and improve severe accident management capabilities.

This report examines the post-accident environmental conditions that could affect the performance of essential instrumentation systems that guide severe accident management involving a core melt and reactor vessel breach. The MELCOR severe accident code was used to examine severe accident scenarios for a BWR/4 with a Mark I containment. The results of this analysis were used as a basis for estimating environmental conditions in the reactor primary containment and reactor building during these scenarios. The conclusion of this study is that given accident scenarios (short-term and long-term blackouts) at a reactor of BWR/4-Mark I design, the performance of critical instrumentation would be considered suspect because environmental qualification values were exceeded.

1. INTRODUCTION

Oak Ridge National Laboratory (ORNL) completed a research activity recently [1] that examined the performance of boiling water reactor (BWR) instrumentation and control (I&C) systems during the days immediately following the Fukushima Daiichi accident, which began on March 11, 2011. A report was issued describing the significant loss of monitoring and I&C systems as a result of the accident, initially due to loss of power. A second report, based on the first report plus prior research on instrumentation used to respond to severe accidents, was also issued [2] which examined the key parameters that instrumentation needs to measure to help operators respond to severe accidents. This report presents the results of an analysis of the environmental conditions that instrumentation may be subjected to during severe accidents using the MELCOR severe accident code [3] to simulate severe accident scenarios. Two severe accident scenarios for a BWR/4 with a Mark I containment, a short-term station blackout (STSBO) and a long-term station blackout (LTSBO), were studied as part of the State-of-the-Art Reactor Consequence Analysis (SOARCA) project sponsored by the U.S. Nuclear Regulatory Commission (NRC) to improve reactor accident analysis efforts [4], [5]. Both of these scenarios involve a loss of all alternating current (ac) power. The STSBO also involves a concurrent loss of direct current (dc) control power, such as during a seismic event. The LTSBO also considers a loss of dc power as battery power is exhausted during the hours following the loss of ac power.

The results of this study served as a basis for estimating environmental conditions in the reactor primary containment building and reactor building during such accidents. Instrumentation that provides essential measurement parameters and values necessary for informing accident management and supporting mitigation efforts experiences these environmental conditions. This report examines the post-accident environmental conditions that could affect the performance of essential instrumentation systems that guide severe accident management involving a core melt and reactor vessel breach.

Section 2 of this report provides background information on instrumentation deficiencies observed as a result of the severe accident conditions affecting Fukushima Daiichi Units 1–3. Section 2 also provides a brief summary of nuclear plant design considerations, including severe accidents, and a discussion of accident monitoring instrumentation requirements and notes severe accident instrumentation parameters. Section 3 summarizes elements of MELCOR severe accident models for the Peach Bottom Atomic Power Station (a BWR/4-Mark I design) as they relate to estimating the environmental conditions in the primary containment and reactor building during potential severe accident conditions. Section 4 concludes the report.

2. BACKGROUND

During the Fukushima Daiichi accident, numerous instrumentation measurements were unavailable as a result of loss of power supplies. Even after power was restored, the instrumentation measurements were inaccurate or differed in values and/or trends across instruments measuring the same parameters. The operating environmental conditions of the instruments certainly affected their performance. For example, Fig. 1 shows different reactor water level readings in redundant instruments A and B in the March 12 time frame for Fukushima Daiichi Unit 1 [6]. For several days later, there were no recorded water level readings from instrument A. Subsequent results were quite different from either measured value, and the indicated values were much higher than level estimates from later analyses. Figure 2 shows reactor pressure readings for two redundant instruments, A and B, for Unit 1 [6]. Pressure readings are missing initially for instrument A. Then, readings for the two instruments show opposite trends before they converge between March 16 and 26. The readings then diverge starting on March 26. The reason for the differences in readings for the two instruments was not provided. As they were received, these readings were given considerable attention by operators as they tried to discern whether significant changes were taking place and to integrate complementary data to verify the readings.

Temperature and pressure conditions in containment affected reactor level and pressure instrumentation and may have complicated accident management. Harsh conditions, including high radiation, high temperature, water spray, and flooding, also likely impacted the performance of other instrumentation, including sensing elements, reference legs, leads, cabling, electronics, transmitters, signal processors, etc., and further challenged accident management and mitigation activities.

The ORNL Fukushima report [1] and the prior report on key parameters for operator diagnosis of plant conditions [2] are part of continuing research efforts to identify and understand the performance of instrumentation used to monitor plant conditions, guide operator response to events, and inform researchers in an effort to improve instrument performance in the challenging environment associated with severe accidents. The ORNL Fukushima report provides information on specific instrumentation issues and references to many readily available detailed accounts of the accident. A parallel study on pressurized water reactor (PWR) instrumentation performance during the Three Mile Island (TMI) Unit 2 accident in 1979 was also conducted [7].

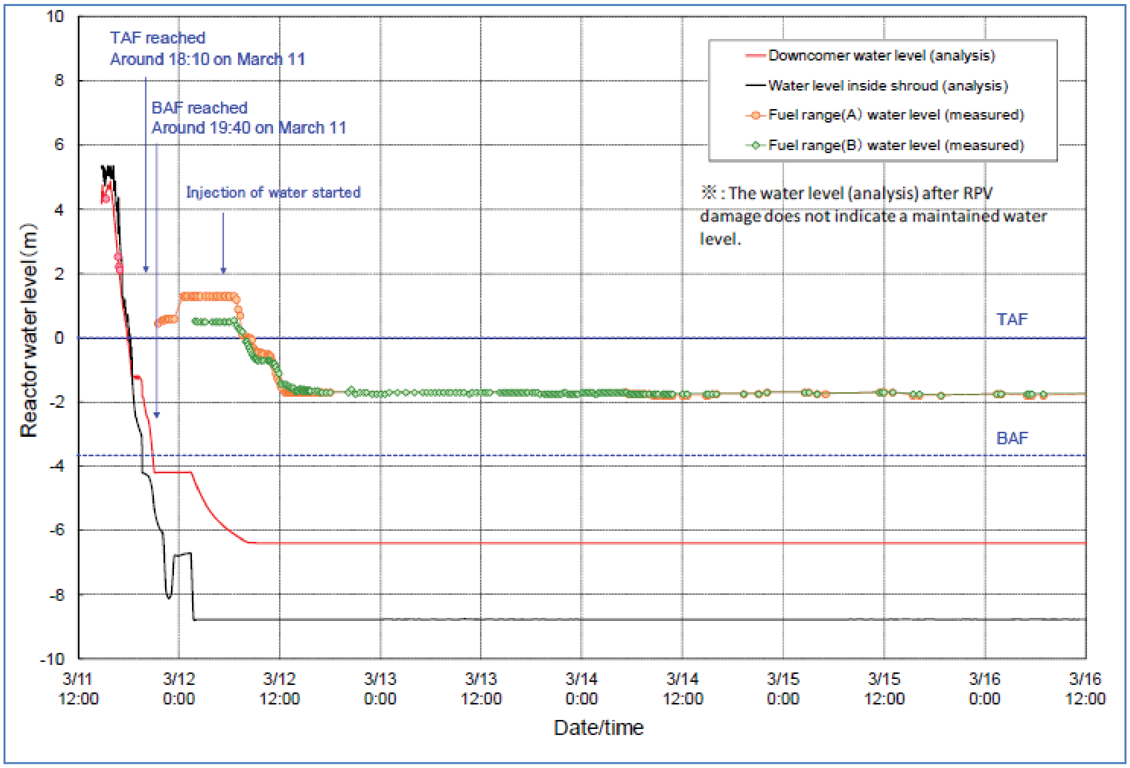


Fig. 1. Actual vs. calculated reactor pressure vessel water level—Fukushima Daiichi Unit 1.

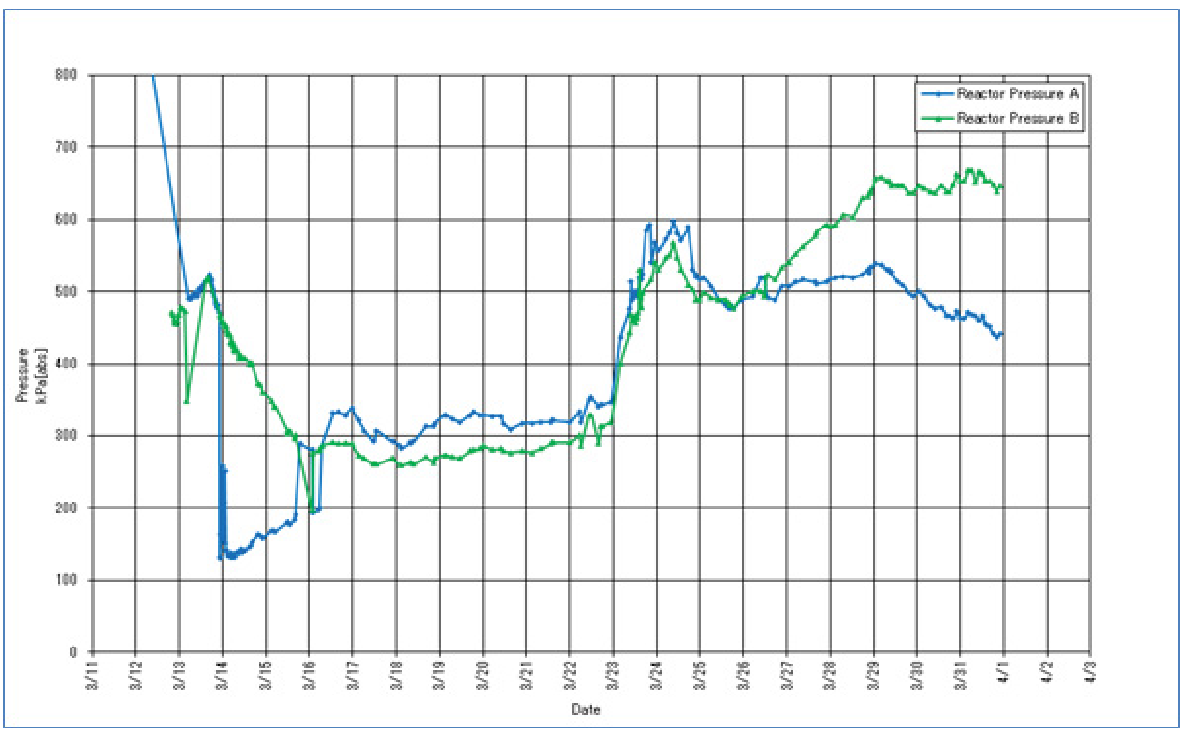


Fig. 2. Reactor pressure vessel—Unit 1.

Based on the ORNL review of the Fukushima Daiichi accident [1], instrumentation that had been monitoring plant parameters at Fukushima Daiichi survived the earthquake and the resulting loss of offsite power systems. However, flooding from the tsunami damaged the onsite power systems for the six-unit site that were powering plant systems as designed following the loss of offsite power caused by the earthquake. This caused the loss of much instrumentation. At the time of the earthquake, Units 1–3 were operating and Units 4–6 were shut down. Core damage occurred within hours at Unit 1 and within days at Units 2 and 3. Hydrogen generated from chemical reactions as the fuel failed at Units 1–3 was released from the reactor vessels into the primary containments; at Units 1–4, hydrogen was released into the reactor buildings. (The Unit 4 reactor building was interconnected with Unit 3 through gas ventilation system piping.) Resulting deflagrations damaged reactor buildings and other structures, systems, and components of Units 1, 3, and 4. Unit 2 reactor vessel integrity and containment integrity were also lost. An air-cooled diesel generator from Unit 6 survived the flooding and was configured to Units 5 and 6 following the tsunami to power critical systems. This prevented damage to fuel at those units.

The dc power systems and electrical rooms for Units 1 and 2 were flooded and lost power when the tsunami waves arrived. Unit 3 retained dc power system function until the batteries were exhausted. Measurement, monitoring, and communications systems were lost or degraded due to the loss of power. As power was restored by temporary batteries gathered by operators, and later by more permanent sources, some instrumentation was able to be repowered. Remote measurement of many parameters was required due to problems repowering systems and control rooms.

As the accidents progressed, the performance of reactor instrumentation systems was affected not only by loss of power but also by the environmental conditions of the accident, including conditions following reactor pressure vessel (RPV) depressurization; containment pressure increases and pressure spikes from hydrogen deflagrations; and high temperatures, which affected RPV pressure and water level measurements. Core damage and RPV damage also released high levels of radioactivity, which can affect the performance of electronic systems, into containment. The impacts of these factors—high temperature, pressure pulses and spikes, high radiation doses, and water spray and flooding conditions— affect instrumentation performance. Thus, there is no continuous or complete record of the severe accident environmental conditions experienced by plant instrumentation systems.

2.1 SEVERE ACCIDENT DESIGN CONSIDERATIONS

Nuclear reactors are designed to protect the public from harm associated with accidents by developing robust designs that accommodate normal operating conditions and less frequent but more serious challenges.

Designers provide systems that can respond to these challenges. Some challenges are normal operating conditions that are anticipated to occur frequently over the lifetime of the plant. Other challenges, termed anticipated operational occurrences, are expected to occur less frequently, up to a few times over the life of the plant.

More serious challenges, design basis accidents, might not be expected to occur over the lifetime of the plant but could occur at a frequency high enough (e.g., less than 10^{-4} to 10^{-5} per year of reactor operation) and with consequences serious enough that they must be considered by the plant designers to ensure that accident consequences are within regulatory limits if they occur.

Another class of accidents—severe accidents, or beyond design basis accidents—could have serious consequences, but the expected frequency of occurrence is below a low threshold (e.g., less than 10^{-5} to 10^{-6} per year of reactor operation for typical current plants and less than 10^{-6} to 10^{-7} per year for more advanced designs).

Severe accidents are considered to be so unlikely that plant designers do not have to design the plant to withstand them, but evaluation of severe accident scenarios does inform plant design. Reactor licensees are subject to requirements associated with severe accidents, such as TMI Action Plan Requirements [8], responses to the review of the Fukushima accident [9], combustible gas control [10], risk from anticipated transients without scram [11], station blackout [12], and fire [13]. Therefore, even though severe accidents are of extremely low probability and not considered credible during a plant's lifetime, they can have significant consequences and merit consideration in establishing a plant's design.

The progression of the accident sequences at Fukushima Daiichi Units 1–3 occurred very much as predicted by prior severe accident research conducted in the United States [14] and internationally, given the complete station blackout that occurred when the tsunami generated by one of the most powerful earthquakes ever recorded rolled on shore. Research noted the importance of I&C and monitoring systems and their dependence on dc electrical power systems when ac electrical systems fail. For example, research in the early 1990s [15] addressed BWR instrumentation availability during severe accidents. Vulnerabilities of plant monitoring systems to various severe accidents were reviewed, and some changes were made. At Fukushima Daiichi Units 2, 4, and 6, air-cooled diesel generators were added as part of the Tokyo Electric Power Company's (TEPCO's) accident management initiatives [16]. The generators provided redundancy and diversity to emergency power sources. However, because of flooded electrical panels, they could not be connected for use at Units 1–4.

2.2 ACCIDENT MONITORING INSTRUMENTATION REQUIREMENTS

2.2.1 Accident Monitoring Design Requirements

Sound design practice and regulatory requirements such as 10 CFR 50, *Domestic Licensing of Production and Utilization Facilities*, for traditional two-part construction and operating licenses and 10 CFR 52, *Licenses, Certifications, and Approvals for Nuclear Power Plants,* for standard design certifications and combined licenses which address functional and design requirements for accident monitoring instrumentation. Regulatory guides and standards address performance criteria such as range, accuracy, response time, operating time, reliability, and design criteria such as independence and separation, isolation, calibration, testability, maintenance, and repair. In the United States, for example, Criterion 13, "Instrumentation and Control," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 [17] requires that "Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions…." Criterion 19, "Control Room," requires reactor licensees to provide a control room "from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions…." Criterion 64, "Monitoring Radioactive Releases," requires reactor licensees "to provide a means for monitoring the primary containment atmosphere, spaces containing components to recirculate loss-of-coolant accident (LOCA) fluids, effluent discharge paths, and the plant environs for radioactivity that may be released as a result of postulated accidents." The terms "accident conditions" and "postulated accidents" refer to design basis accidents in this context.

Subsection (2)(xix) of 10 CFR 50.34(f), "Additional TMI-Related Requirements," requires that licensees provide "instrumentation adequate for monitoring plant conditions following an accident that includes core damage."

NRC Regulatory Guide 1.97 addresses accident monitoring instrumentation. NRC Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions during and following an Accident," dated May 1983 [18], provides a prescriptive approach to the design and qualification criteria for instrumentation and lists variables to be monitored. Revision 3 has been widely followed by the U.S. nuclear fleet. Regulatory Guide 1.97, Revision 4, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," dated June 2006 [19], is intended for licensees of new nuclear power plants (i.e., those plants licensed following the issuance of Revision 4). Revision 4 does not supersede previous revisions for licensees of currently operating reactors. Revision 4 endorses (with exceptions) industry standard IEEE STD 497-2002 [20], *IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations*, promulgated by the Institute of Electrical and Electronic Engineers (IEEE). Instead of the prescriptive nature of Revision 3, Revision 4 provides more flexible, performance-based criteria for use in selecting variables based on the accident management functions of the given type of variable.

2.2.2 Accident Monitoring Instrumentation Variables for BWRs

Accident monitoring instrumentation covers a spectrum of variables that support accident management needs and requirements. One class of variables is to inform manual actions to be taken by control room operators when no automatic controls are provided but are necessary for accomplishing safety functions for design basis events. Other classes show that safety functions such as reactivity control, core cooling, reactor coolant system integrity, and containment are being accomplished; indicate actual or potential breach of fission product release barriers; monitor the condition and performance of individual systems important to safety; and determine the magnitude of radioactive releases.

NRC Regulatory Guide 1.97, Revision 3 [18], provides a specific list of instrument variables to monitor for PWRs and BWRs. Given that the BWR/4-Mark I plants are mature designs and were in use at the time Revision 3 was issued, the variables are highlighted here as typical of those needed for accident monitoring, recognizing that this guidance remains in effect for licensees of currently operating plants. Variables to be monitored were broken into several categories, including the following.

- Reactivity control—neutron flux, control rod position, soluble boron concentration (grab sample)

- Core cooling—reactor vessel water level, reactor core isolation cooling system flow, high-pressure coolant injection flow, low-pressure coolant injection flow, core sprays system flow, standby liquid control system flow, residual heat removal system flow, and residual heat removal system heat exchanger outlet temperature. Variables associated with fuel cladding are radioactivity concentration or radiation level in the circulating reactor coolant and gamma spectrum analysis of reactor coolant.

- Reactor coolant system pressure boundary integrity—reactor coolant system pressure, drywell pressure, drywell sump level, primary containment area radiation, and suppression pool water level

- Containment integrity and containment radiation—primary containment pressure, suppression pool water level, drywell atmosphere temperature, drywell spray flow, primary containment isolation valve position (closed/not closed—excluding check valves), drywell hydrogen concentration, containment effluent radioactivity from release points, including standby gas treatment vent, effluent radioactivity from buildings or areas where penetrations or hatches in direct contact with primary containment are located, primary containment area high radiation, and reactor building or secondary containment area radiation

- Condensate and feedwater system—main feedwater flow and condensate storage tank level

- Main steam system—main steam line isolation valve leakage control system pressure (if the system is provided) and reactor coolant system pressure relief valve and automatic depressurization system valve positions

- Cooling water system— cooling water temperature to engineered safety feature system components and cooling water flow to engineered safety feature system components

- Radioactive waste systems— high-radioactivity liquid tank level

- Ventilation systems— emergency ventilation system damper position

- Power supplies and other energy sources— Plant-specific variables (e.g., voltages, currents, pressures) associated with the status of standby power and other energy sources important to safety (e.g., electric, hydraulic, and pneumatic)

- Area radiation— radiation exposure rate inside buildings or areas where access is required to service equipment important to safety

- Airborne radioactive materials released from plant—drywell purge and standby gas treatment purge, secondary containment purge, auxiliary buildings containing reactor coolant system gases such as decay tanks, and common plant vent. Variables are associated with monitoring airborne radioactive particulates or halogens from all identified plant release points.

- Environs radiation and radioactivity— airborne radiohalogens and particulates (portable sampling), plant and environs radiation (portable instrumentation)

- Meteorology—wind direction, wind speed, and estimation of atmospheric stability based on the vertical temperature difference from the primary meteorological system

- Accident sampling of reactor coolant system and sumps and containment air sampling

Reactor vessel temperature is not listed in NRC Regulatory Guide 1.97, Revision 3; however, it is an important parameter for accident management purposes.

2.2.3 Key Variables and Parameters

All of the variables noted in Section 2.2 are important from an accident-monitoring perspective. But for the purpose of this report, it is assumed that core damage has already occurred (i.e., fuel rod clad rupture and release of fission products into the reactor system) and that reactor protection systems have already been actuated to take the reactor subcritical. Key information identified in [2] needed to prevent or delay core dispersal from the reactor vessel, maintain containment integrity, and mitigate fission product release focused on the conditions of the reactor pressure vessel, drywell, suppression pool, and reactor building is summarized in Table 1. Note that some of the information needs may not be directly provided. For example, drywell water level may be inferred from other data, such as drywell pressure.

Portable instruments that could be used to measure releases within and external to the plant environs are not within the scope of this report. Additionally, because of attention already placed on improved spent-fuel pool monitoring instrumentation (e.g., NRC Order EA-12-051 issued March 12, 2012, effective on issuance for licensees to modify licenses with regard to reliable spent-fuel instrumentation [21]), spent-fuel pool monitoring instrumentation is not addressed in this report.

Table 1. Summary of key severe accident information needs

Reactor fuel temperature (no direct measurement)
Reactor vessel water level
Reactor vessel pressure
Reactor vessel water injection flow rates
Reactor vessel pressure relief valve position indication
Reactor vessel temperature
Drywell water level
Drywell pressure
Drywell hydrogen and oxygen concentration
Drywell radiation
Drywell temperature
Drywell spray flow rate
Containment isolation valve position indication (for reactor coolant pressure boundary verification and containment integrity assurance)
Suppression pool water level
Suppression pool temperature
Suppression pool pressure
Suppression pool spray flow rate
Suppression pool gas space hydrogen and oxygen concentration
Reactor building area radiation
Reactor building area temperature
Reactor building differential pressure
Ventilation and exhaust radiation
Sump or room water level

2.2.4 Instrumentation Location Information

After plant accident monitoring instrumentation needs are determined, including the key instrumentation highlighted above, specifications for the instruments would be defined. These include the parameter range, accuracy requirements, operating lifetime and environmental conditions (e.g., temperatures, pressures, fluids, corrosive conditions, radiation levels, vibration, etc.) expected for the harsh conditions of potential design basis accidents—with appropriate margin to account for various uncertainties and potential unknown issues and be consistent with regulatory requirements. These factors depend on the locations of the instruments and the necessary subcomponents and support elements such as sensors, power and control cables, transducers, transmitters, reference and sensing lines, cabinets/racks, and indicators.

Margin in instrumentation designs may enable them to function in some capacity in severe accident conditions for a period of time beyond the requirements for design basis accidents. Severe accident assessment models may provide improved estimates of environmental conditions for various instrument locations so that improvements in survivability and performance can be made, if needed, thereby improving accident management and mitigation capability.

3. MELCOR SEVERE ACCIDENT MODEL OF THE PEACH BOTTOM ATOMIC POWER STATION

The MELCOR severe accident code [3] has been in use for over 30 years to study the phenomenology and progression of severe accidents in which radioactive material escapes from the reactor core, reactor pressure vessel and reactor coolant pressure boundary, and reactor containments to the environment. MELCOR was recently used to estimate environmental conditions in primary containment and the reactor building at the Peach Bottom Atomic Power Station under potential severe accident conditions during severe accident scenarios as described in the *State-of-the-Art Reactor Consequence Analyses Project—Volume 1: Peach Bottom Integrated Analysis* [4]. This study and a similar study for pressurized water reactors were conducted by Sandia National Laboratories and are referred to as the SOARCA study. Models developed for the SOARCA study included refinements and expansions regarding the spatial representation of the primary containment and the reactor building. These refinements help make MELCOR suitable for estimating severe accident conditions in these areas.

3.1 PRIMARY CONTAINMENT AND REACTOR BUILDING CONTROL VOLUMES

The BWR primary containment and reactor building are of principal interest in this study, which is focused on the environmental conditions under which instrumentation systems would have to survive to remain functional during a severe accident. The various instrumentation systems monitor structures, systems, components, and areas of the primary containment or reactor building that have important roles in accident management and mitigation. Instrumentation system components there include sensors, power and control cables, transducers, transmitters, reference and sensing lines, indicators, etc., which may be degraded by harsh conditions of the accident and hinder accident response.

The Peach Bottom MELCOR model includes separate hydrodynamic control volumes and heat structures for the drywell and wetwell. The drywell is represented by four control volumes that allow for variations during late phases of a severe accident. The primary containment nodalization scheme and control volumes, labeled as control volume (CV) 202, etc., are shown in Fig. 3. Two control volumes are used for the wetwell. Control volumes for the reactor building are shown in Figs. 4 and 5 for Sections A-A and B-B, respectively. The control volumes are summarized in Table 2.

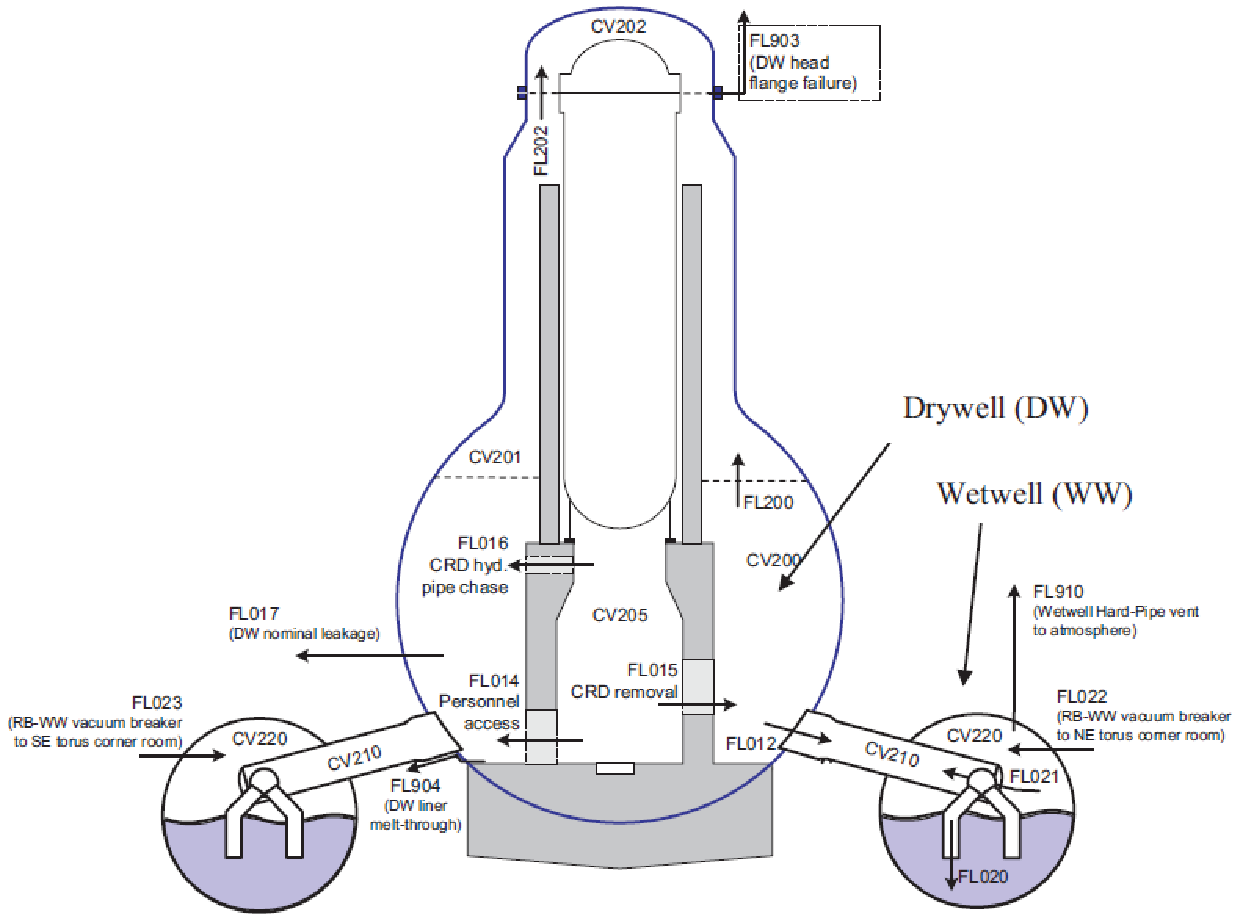


Fig. 3. BWR/4-Mark I primary containment. [4]

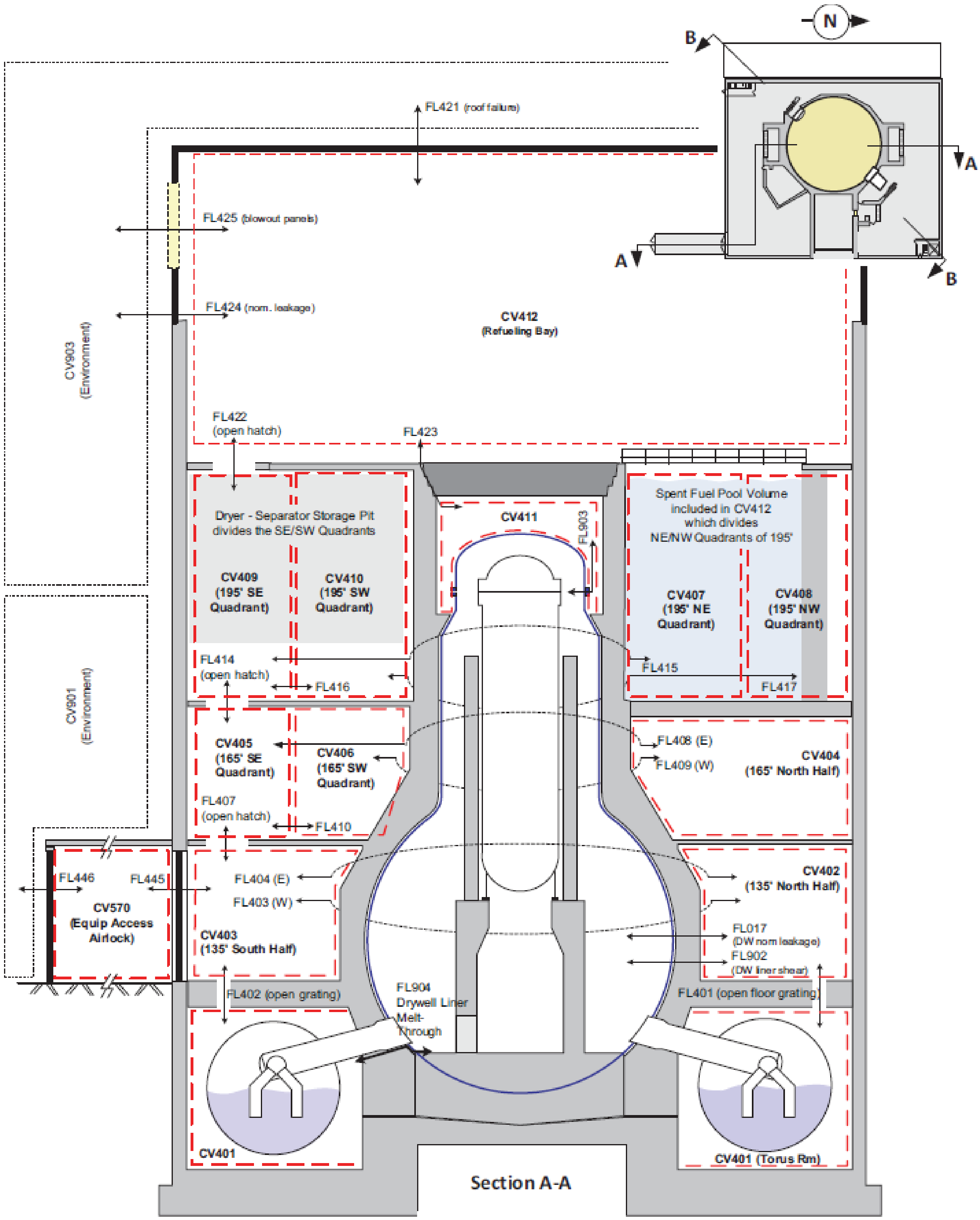


Fig. 4. BWR/4-Mark I reactor building, Section A-A. [4]

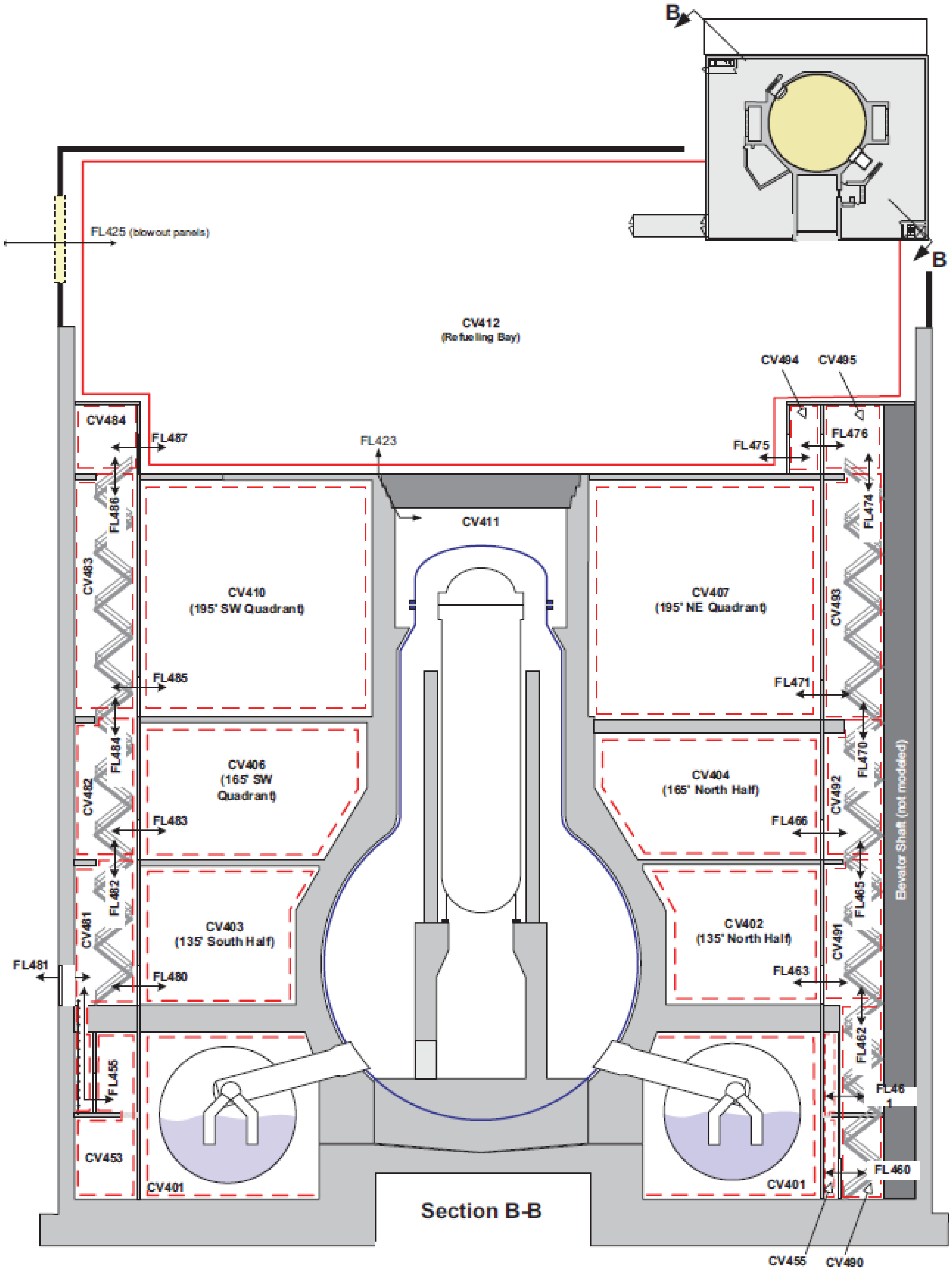


Fig. 5. BWR/4-Mark I reactor building, Section B-B. [4]

Table 2. MELCOR primary containment and reactor building control volumes

Control volume	Description	Control volume	Description
CV 200	Drywell area below reactor vessel	CV 410	Reactor building 195 ft SW quadrant
CV 201	Drywell reactor vessel elevation area	CV 411	Reactor building area above drywell head
CV 202	Drywell head area	CV 412	Refueling bay
CV 205	Drywell reactor vessel pedestal area	CV 453	Reactor building torus room elevation stairwell south half
CV 210	Drywell to wetwell vent	CV 455	Reactor building torus room elevation adjacent to stairwell north half
CV 220	Wetwell/torus	CV 481	Reactor building 135 ft elevation stairwell south half
CV 320	Reactor vessel lower plenum	CV 482	Reactor building 165 ft elevation stairwell south half
CV 401	Reactor building torus room	CV 483	Reactor building 195 ft elevation stairwell south half
CV 402	Reactor building 135 ft north half	CV 484	Reactor building refueling floor elevation stairwell south half
CV 403	Reactor building 135 ft south half	CV 490	Reactor building torus room elevation stairwell north half
CV 404	Reactor building 165 ft north half	CV 491	Reactor building 135 ft elevation stairwell north half
CV 405	Reactor building 165 ft SE quadrant	CV 492	Reactor building 165 ft elevation stairwell north half
CV 406	Reactor building 165 ft SW quadrant	CV 493	Reactor building 195 ft elevation stairwell north half
CV 407	Reactor building 195 ft NE quadrant	CV 494	Reactor building refueling floor elevation adjacent to stairwell north half
CV 408	Reactor building 165 ft NW quadrant	CV 495	Reactor building refueling floor elevation stairwell north half
CV 409	Reactor building 195 ft SE quadrant	CV 570	Reactor building equipment access airlock

3.2 MELCOR ACCIDENT MODELING SCENARIOS

The SOARCA report noted that accident sequences initiated by internal and external events were identified and reviewed. However, internal sequences with an occurrence frequency above 10^{-6} per reactor year were found not to result in core damage (or 10^{-7} per reactor year for certain sequences that could generate significant early releases of radionuclides or involve a containment bypass pathway) [4]. External event scenarios (i.e., those initiated by internal flooding and fire, seismic events, extreme wind, etc.) were developed.

Of these, seismic-initiated events were determined to be the most restrictive due to the timing of equipment failures and challenges for mitigation and protective actions and were chosen as representative of events with a significant magnitude and effect on important plant systems. The seismic events result in the loss of offsite and onsite ac power that were considered in two scenarios, an STSBO and LTSBO, both of which involve a loss of all ac power. The STSBO also involves a concurrent loss of dc control power, such as during a seismic event. The LTSBO also involves a loss of dc power as battery power is exhausted during the hours following the loss of ac power.

Possible mitigation measures to respond to the seismic events were incorporated into the scenarios. However, for purposes of this report, the successful use of mitigation measures implemented following the events of September 11, 2001, pursuant to 10 CFR 50.54(hh) was not incorporated. The reason for this was to better reflect the conditions present in the Fukushima Daiichi accident in which these mitigative measures (or the Japanese mitigation options) could not be implemented. Thus, the unmitigated LTSBO and STSBO scenarios were selected for analysis.

3.3 MODELING LIMITATIONS

The MELCOR SOARCA models were developed and used to provide more detailed, integrated, and realistic analyses of the consequences of severe accidents at commercial nuclear power plants. These include the estimation of accident source terms, progression, and offsite consequences. The models were not designed to provide environmental conditions within the primary containment and reactor building with a high degree of specificity and precision—a fine nodalization schema (many nodes) in these structures was not necessary to sufficiently define the drivers for release of portions of the accident source term to the environment. Therefore, gradients in environmental conditions within the fairly large nodes of the primary containment and the reactor building are not determined. For some variables, such as pressure, gradients might be small. For others, such as temperature or fission product inventory, gradients could be larger. MELCOR also does not have the provision for estimating the conditions at specific component locations which could be useful for assessing the various effects, such as heat radiated to an instrumentation component from an adjacent high-temperature component.

Although an important component of the MELCOR model is the time-dependent fission product inventory in the various model nodes, MELCOR does not calculate radiation dose. However, Sandia modeling staff developed an algorithm for estimating time-dependent radiation dose in a node based on the fission product inventory in the node. A limitation of this algorithm is that it does not include the contribution to dose from neighboring nodes or the potential effects of a steam environment and structures within the nodes that could provide shielding to certain components, depending on their specific locations within a node. Other limitations associated with the lack of node-to-node communications is that radiation from core debris on the containment floor is not incorporated into the dose calculations for the nodes and the inability to account for radiation from particles deposited on or near components

Given the uncertainties in the magnitude and timing of accident progression and ways in which primary containment and reactor building integrity could be degraded (i.e., hydrogen deflagrations), it is apparent that trying to provide specificity and precision in environmental conditions in numerous nodes over time would be challenging.

3.4 MODEL RESULTS

Staff at Sandia National Laboratories executed the MELCOR SOARCA model of the Peach Bottom Power Atomic Power Station for the postulated unmitigated STSBO and LTSBO scenarios and provided time-dependent thermal hydraulic and radiation doses for control volumes in the reactor drywell, wetwell, and reactor building. Graphs of the various parameters are shown in the figures in Appendixes A and B for the STSBO and LTSBO cases, respectively. The maximum values of the thermal hydraulic parameters and the duration of these conditions are important while cumulative radiation dose is important. Typically, environmental conditions associated with design basis accidents form the bases for individual plant equipment functional requirements and procurement specifications.

This study bears two limitations: Peach Bottom models were used for the MELCOR analyses. Precise equipment locations were not identified, partly because elements of many instrumentation components, such as sensors, power and control cables, transducers, transmitters, reference and sensing lines, indicators, etc., may traverse through multiple control volumes. Second, the procurement specifications for the Peach Bottom equipment were not requested, nor were specific equipment survivability threats directly assessed due to the reduced rigor of this assessment. Even if this information had been provided, conclusions based on the Peach Bottom equipment and its location, and the Peach Bottom-specific MELCOR analyses, would not be directly applicable to other BWRs, even those of similar design.

However, the results from these analyses of several environment conditions and cumulative radiation doses for the control volumes used in the Peach Bottom MELCOR SOARCA study may be representative of the effects of the environmental conditions of similar designs. Environmental conditions that threaten the often dispersed elements of I&C systems could threaten the systems.

3.4.1 Maximum Pressure and Temperature Values

Pressure and temperature conditions in containment and reactor building areas were predicted during the initial 48 h for the STSBO and LTSBO scenarios. Table 3 shows the maximum values for pressure and temperature for the control volumes during the initial 48 h period of the STSBO scenario and the cumulative beta and gamma radiation doses for the 1 year period. Table 4 shows the maximum values for pressure and temperature for the control volumes during the initial 48 h period of the LTSBO scenario and the cumulative beta and gamma radiation doses for the 1 year period.

Maximum pressure values in the drywell and wetwell were approximately 88 psia in the STSBO and 100 psia in the LTSBO. The timeframe of maximum pressures correlates with periods of high hydrogen concentration and can be seen when the pressure charts are compared with the hydrogen mole fractions for the STSBO scenario, as illustrated in Figs. 6 and 7. The phenomenology of these severe accident scenarios is complex and an explanation of the difference between the STSBO and LTSBO scenarios would take a considerable effort beyond the scope of this study. However, ORNL and SNL staff discussed possible factors affecting the difference. For example, in the LTSBO case there is a greater amount of decay heat deposited in containment (due to the reactor core isolation cooling system [RCIC] running and rejecting heat to containment) which could raise the base containment pressure prior to the pressure spikes. Additionally, while RCIC ran, the water it injected from the condensate storage tank would have reduced the free containment volume which could have concentrated the effect of the pressure spikes. The timing of containment failure could also affect maximum pressure values in the scenarios. Ultimately, a thorough review of the model parameters and results would be required to explain the difference between the two scenarios.

Maximum temperature values of the drywell were approximately 1929 K (3013°F) in the STSBO and 1722 K (2640°F) in the LTSBO in control volume (CV) 205 and, generally, the maximum temperature values in the containment and reactor building control volumes were higher in the STSBO scenario. The temperature in the drywell in the STSBO scenario is shown in Fig. 8, which is illustrative of the other control volumes. It was observed that the STSBO scenario has about a 10% higher mole fraction of hydrogen in the drywell, possibly providing a greater source of energy for deflagration and higher temperatures for that scenario. A detailed review of the model parameters and results would be required to be sure of the difference.

3.4.2 Duration of High Pressure and Temperature Conditions

The duration of high pressures and temperatures, as well as the magnitude, are important factors in qualification of instruments for harsh environments. Table 3 of NUREG/CR-5444 [15] lists instrument qualification conditions as referenced to the Peach Bottom Final Safety Analysis Report (FSAR). Conditions for non-specific primary containment and reactor building locations are summarized as follows:

Primary containment: Pressure, 49 psig; Temperature, 317°F; Radiation dose, 3.5-4.4E+07 rad

Reactor building: Pressure, 0-2 psig; Temperature, 120-250°F; Radiation dose, 3.5E+04 rad

Charts of primary containment and reactor building pressure and temperature values during the first 48 h of an STSBO scenario are shown in Fig. **6**. The points to take from these charts are the values of the parameter relative to the time above the qualification values. For example, for the drywell and wetwell, the pressure exceeds the instrument qualification value for less than 2 h while the atmospheric temperature exceeds the instrument qualification value for approximately 40 h—the duration of the transient shown in the chart. For the reactor building, there is only a brief period where the pressure in one reactor building control volume exceeds the instrument qualification range while the temperature exceeds or is very near the instrument qualification range for numerous control volumes for approximately 40 h.

A similar chart was prepared to show the primary containment and reactor building pressure and temperature values during the first 48 h of an LTSBO scenario as shown in Fig. **7**. These charts are similar to the results of the STSBO. For the drywell and wetwell, the pressure exceeds the instrument qualification value several hours while the drywell and wetwell atmospheric temperature exceeds or is near the instrument qualification value for approximately 30 h—the duration of the transient shown in the chart. For the reactor building, the pressure does not exceed the instrument qualification range while the temperature exceeds or is very near the instrument qualification range for numerous control volumes for approximately 40 h.

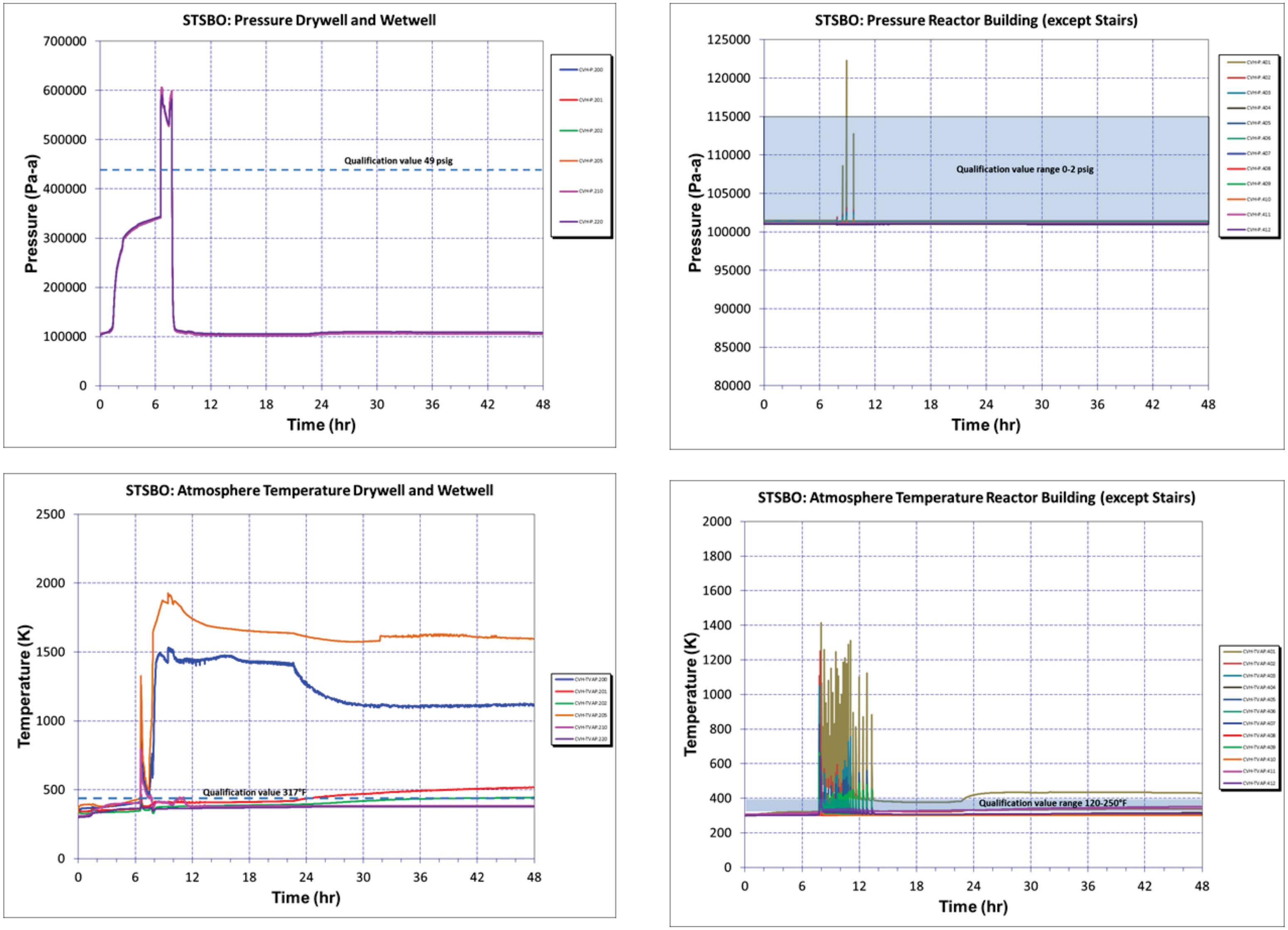


Fig. 6. STSBO pressures and temperatures vs. qualification values.

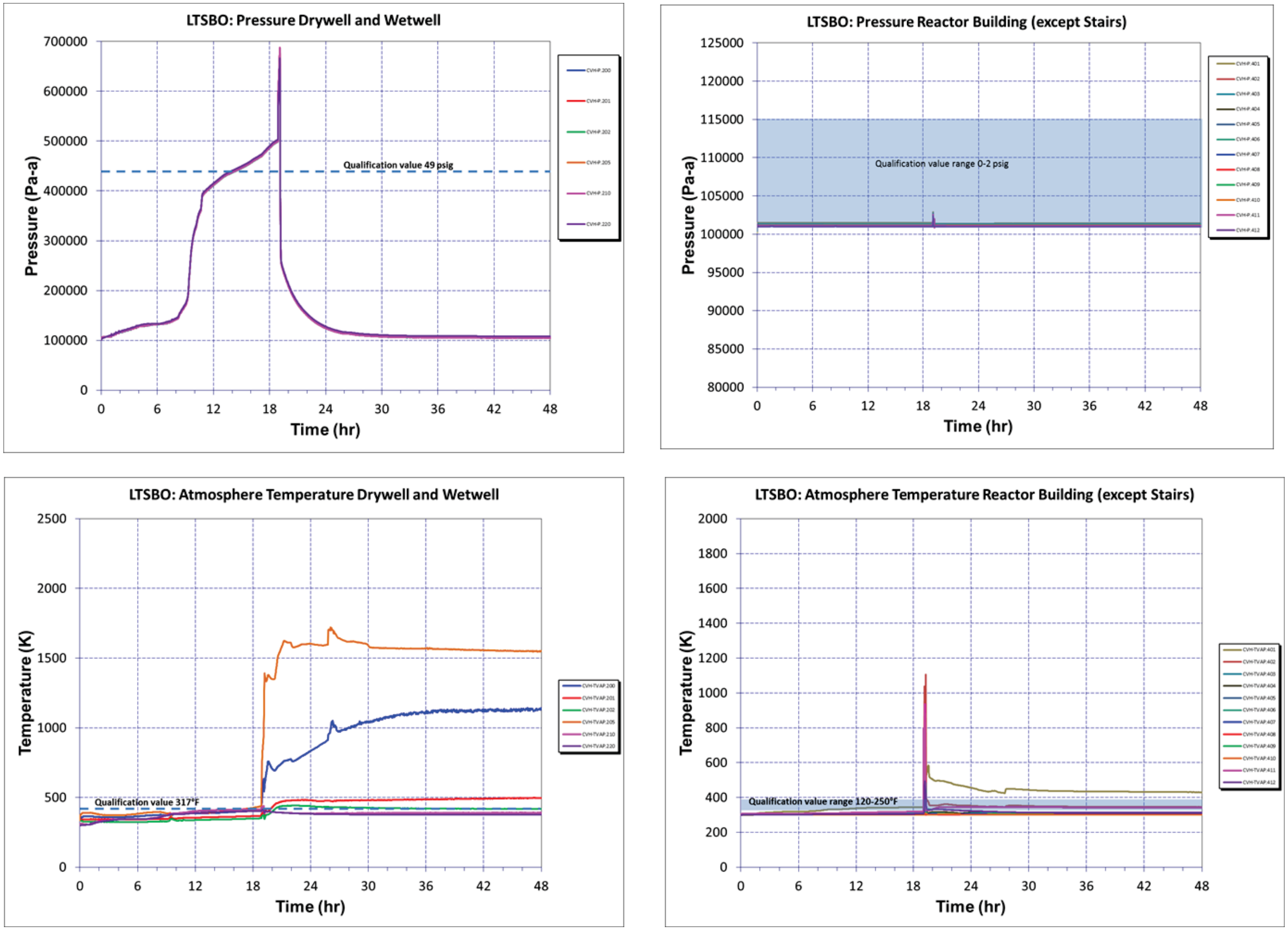


Fig. 7. LTSBO pressures and temperatures vs. qualification values.

Table 3. Maximum values for parameters per control volume (STSBO)

Area	Control Volume	Pressure (Pa abs)	Pressure (psia)	Temperature (K)	Temperature (°F)	Beta Radiation Dose (Rad)	Gamma Radiation Dose (Rad)
Drywell	CV 200	606256	87.9	1534	2302	5.63E+07	5.53E+07
	CV 201	605980	87.9	518	473	4.52E+07	1.68E+07
	CV 202	605567	87.8	444	340	3.61E+07	9.25E+06
	CV 205	606256	87.9	1929	3013	7.17E+07	2.88E+08
	CV 210	606325	87.9	792	966	6.95E+09	4.17E+08
Wetwell	CV 220	589778	85.5	392	246	6.28E+10	5.49E+09
Torus room	CV 401	122244	17.7	1413	2084	7.31E+07	4.86E+07
Reactor building	CV 402	103146	15.0	1252	1794	2.00E+06	6.92E+06
	CV 403	102525	14.9	1049	1429	5.16E+05	2.80E+06
	CV 404	102180	14.8	387	237	6.87E+05	2.12E+06
	CV 405	102180	14.8	827	1029	4.86E+05	1.16E+06
	CV 406	102180	14.8	537	507	4.65E+05	2.29E+06
	CV 407	101629	14.7	320	116	4.30E+06	2.22E+07
	CV 408	101629	14.7	303	86	1.57E+07	1.76E+07
	CV 409	101629	14.7	663	734	8.02E+06	6.60E+06
	CV 410	101629	14.7	326	127	1.64E+07	1.13E+07
	CV 411	101215	14.7	352	174	4.35E+05	2.60E+03
	CV 412	101215	14.7	642	696	2.26E+06	1.84E+06
	CV 453	101698	14.8	300	80	3.37E+05	5.40E+06
	CV 455	112453	16.3	1824	2824	2.42E+07	1.81E+07
Stairwells	CV 481	102249	14.8	397	255	8.33E+05	1.31E+08
	CV 482	102180	14.8	323	122	1.11E+04	8.91E+05
	CV 483	102042	14.8	314	106	1.07E+05	3.23E+04
	CV 484	101905	14.8	375	215	2.35E+05	8.16E+04
	CV 490	103215	15.0	766	919	8.88E+06	2.38E+07
	CV 491	102732	14.9	391	244	7.03E+05	4.77E+04
	CV 492	102180	14.8	318	113	5.03E+05	5.24E+04
	CV 493	102111	14.8	316	109	5.73E+05	3.96E+05
	CV 494	101215	14.7	337	147	1.71E+06	1.29E+07
	CV 495	101973	14.8	314	106	5.86E+05	4.34E+05
Equipment airlock	CV 570	101973	14.8	694	790	6.61E+03	1.12E+02

Table 4. Maximum values for parameters per control volume (LTSBO)

Area	Control Volume	Pressure (Pa abs)	Pressure (psia)	Temperature (K)	Temperature (°F)	Beta Radiation Dose (Rad)	Gamma Radiation Dose (Rad)
Drywell	CV 200	687407	99.7	1142	1596	4.07E+07	4.0E+06
	CV 201	687407	99.7	498	437	2.18E+07	1.0E+06
	CV 202	686718	99.6	442	336	1.12E+07	5.7E+05
	CV 205	687407	99.7	1722	2640	4.34E+07	9.9E+06
	CV 210	687407	99.7	438	329	6.41E+09	9.4E+08
Wetwell	CV 220	666723	96.7	413	284	5.65E+10	5.0E+09
Torus room	CV 401	102732	14.9	584	592	5.36E+07	3.9E+06
Reactor building	CV 402	102732	14.9	1105	1529	3.69E+05	5.1E+05
	CV 403	102732	14.9	516	469	1.51E+05	6.9E+04
	CV 404	102732	14.9	343	158	2.12E+05	1.3E+05
	CV 405	102732	14.9	642	696	1.35E+05	1.28E+05
	CV 406	102732	14.9	320	116	2.41E+05	1.1E+05
	CV 407	102732	14.9	308	95	5.83E+05	9.1E+04
	CV 408	102732	14.9	302	84	6.41E+05	1.9E+05
	CV 409	102732	14.9	654	718	3.23E+05	6.6E+04
	CV 410	102732	14.9	308	95	3.84E+06	1.6E+05
	CV 411	102732	14.9	936	1225	1.13E+08	1.2E+07
	CV 412	102732	14.9	494	430	4.56E+05	9.4E+04
	CV 453	102042	14.8	300	80	5.33E+05	2.9E+05
	CV 455	102732	14.9	704	808	4.22E+07	1.8E+06
Stairwells	CV 481	102732	14.9	305	89	1.03E+05	2.9E+05
	CV 482	102732	14.9	305	89	1.11E+06	3.1E+04
	CV 483	102042	14.8	314	106	1.74E+06	3.3E+04
	CV 484	102042	14.8	407	273	1.99E+05	1.9E+04
	CV 490	102732	14.9	1230	1754	2.33E+07	1.1E+06
	CV 491	102732	14.9	353	176	1.30E+05	5.7E+04
	CV 492	102042	14.8	312	102	1.22E+05	6.3E+04
	CV 493	102042	14.8	308	95	1.57E+05	5.1E+04
	CV 494	102042	14.8	333	140	1.84E+05	2.0E+05
	CV 495	102042	14.8	318	113	1.36E+05	2.9E+04
Equipment airlock	CV 570	101353	14.7	333	140	3.06E+03	3.2E+01

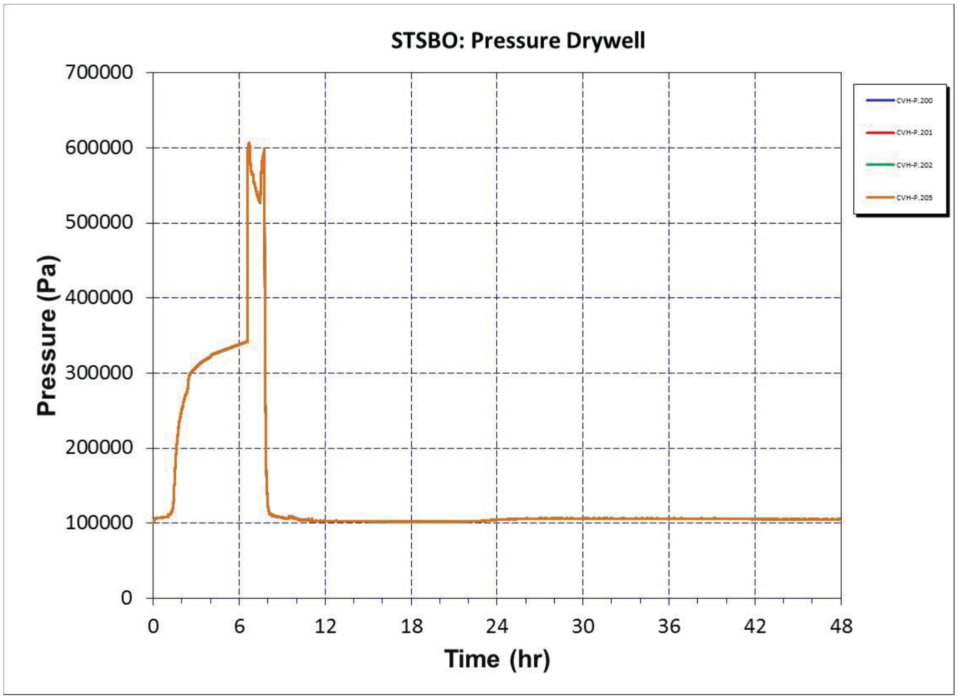


Fig. 8. Pressure for drywell control volumes.

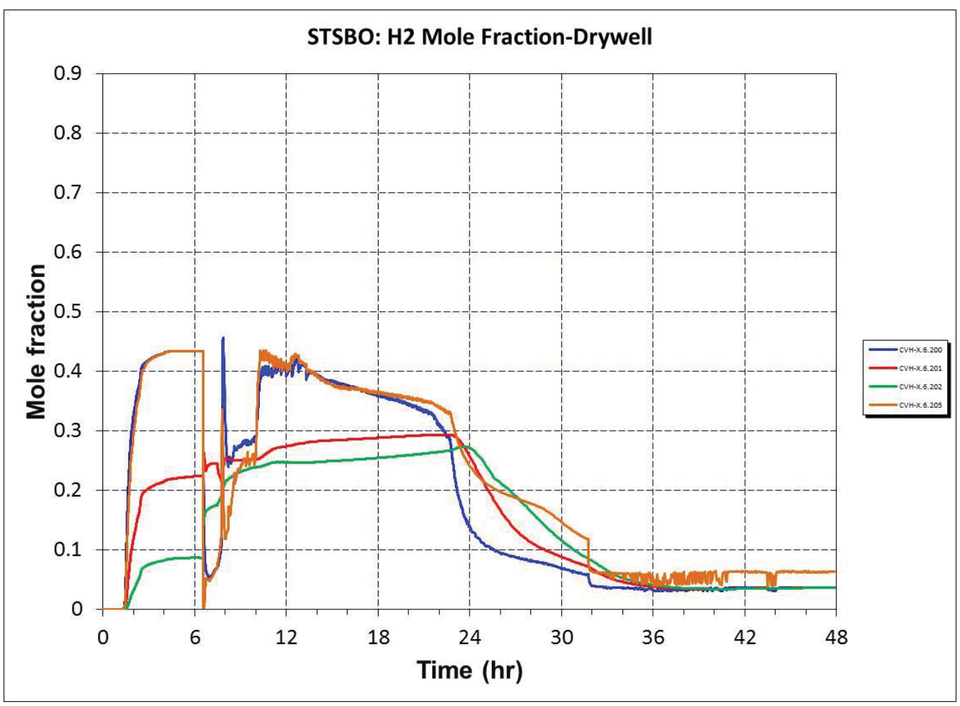


Fig. 9. Hydrogen mole fraction for drywell control volumes.

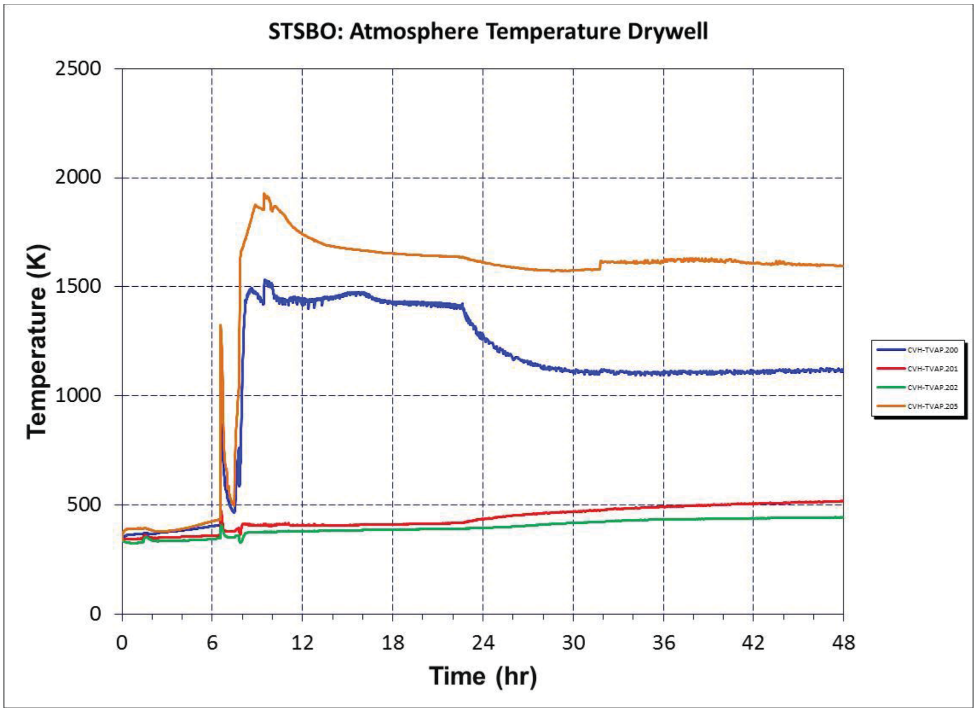


Fig. 10. Temperature for drywell control volumes.

3.4.3 Cumulative Beta and Gamma Radiation Doses

Cumulative beta and gamma radiation doses over a 1 year period are shown in Table 3 for the STSBO scenario. Cumulative beta and gamma radiation doses over a 1 year period are shown in Table 4 for the LTSBO scenario. Figure 9 illustrates the buildup of the cumulative beta doses during the year for the drywell control volumes for the STSBO scenario. Figure 10 illustrates the buildup of the cumulative gamma doses during the year for the drywell control volumes for the STSBO scenario. As can be seen, approximately 90% of the cumulative beta dose for the year occurs in the initial hours of the event, while it takes about 200 days to receive approximately 90% of the cumulative gamma dose for the year. The gamma dose qualification condition of 4.4E+07 rad from Table 3 of NUREG/CR-5444 [15] is also plotted to show how extensively the estimated dose exceeds the qualification condition dose. (Figures showing cumulative radiation doses shown in the appendices may be used to estimate cumulative doses for alternate intervals, if desired.)

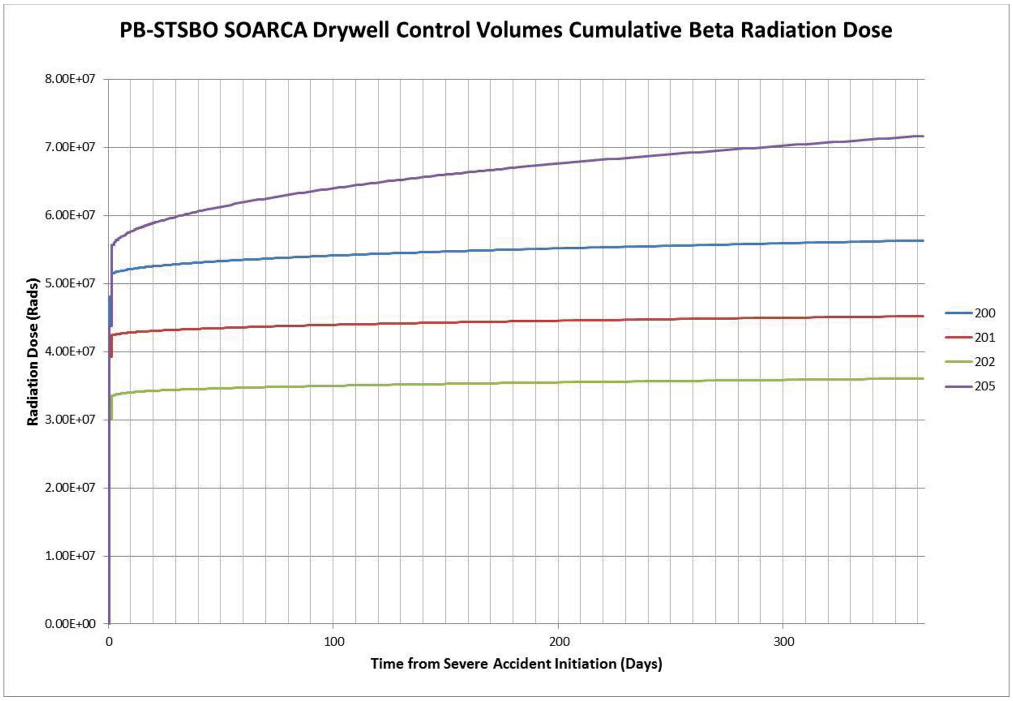


Fig. 11. Cumulative 1 year beta doses for drywell control volumes.

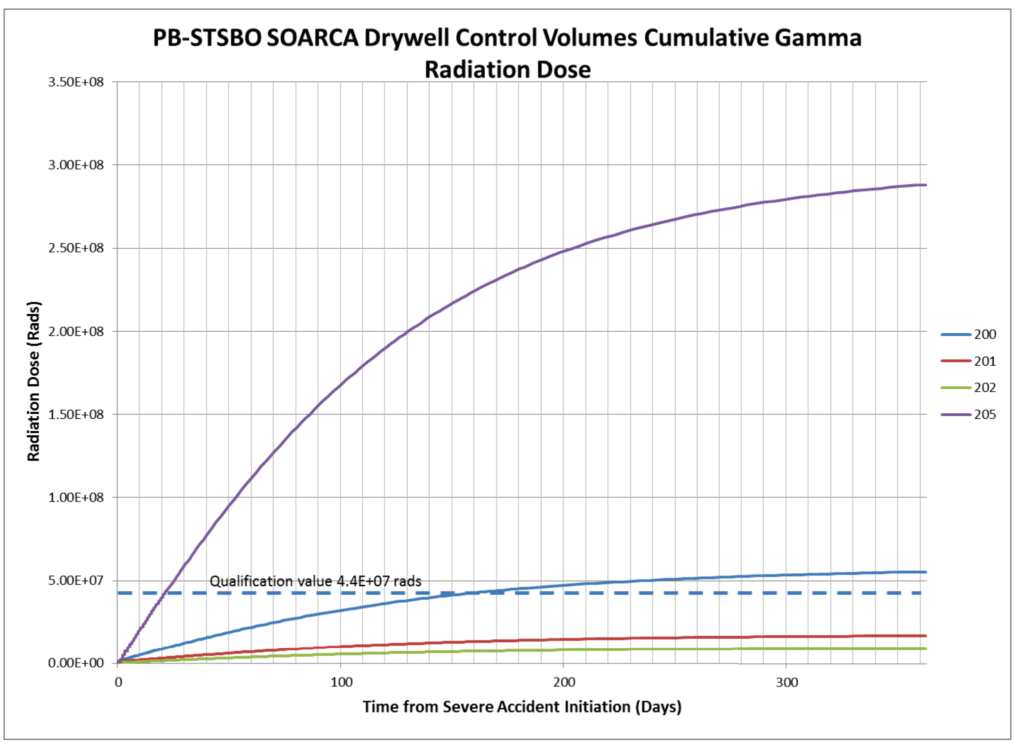


Fig. 12. Cumulative 1 year gamma doses for drywell control volumes.

3.4.4 Significance of Results

The maximum values of the pressure, temperature, and radiation for the control volumes in the drywell and wetwell, reactor building rooms, and reactor building stairwells are shown in Tables 3 and 4. These values were compared with qualification conditions for instruments for selected plant parameters noted in Table 3 of NUREG/CR-5444 [15], as shown in Table 5. The qualification conditions noted in NUREG/CR-5444 were referenced to the Peach Bottom FSAR. NUREG/CR-5444 does not note specifically where the instrumentation is located. It typically references a more general building or area, such as the reactor building or the drywell.

The primary purpose of the table is to illustrate that instrument environmental qualification conditions, especially temperature and/or radiation conditions, were exceeded for instruments located in reactor building and drywell locations. Instruments located in the reactor building will be discussed first.

3.4.4.1 Instruments located in the reactor building

The discussions that follow assume that the instruments located in the reactor building are located in the subset of 10 of the 12 reactor building control volumes: CV 402 through CV 410 plus CV 412. Control volumes for the torus room level (CV 401) and around the drywell head (CV 411) were assumed not to contain the instruments used to monitor the parameters shown in Table 5. (Plant-specific information would be helpful to specify the most appropriate control volumes.)

For five of 15 plant parameters listed in Table 5, instrument qualification pressure conditions were exceeded in five of these 10 reactor building control volumes for the STSBO scenario, although typically by only 0.1 to 0.3 psi. For the LTSBO scenario, instrument qualification pressure conditions for these five parameters were exceeded in all 10 control volumes, although typically by only 0.1 to 0.3 psi. The conditions were also only exceeded for a short interval (spike) of time.

For 14 of the 15 plant parameters listed in Table 5, instrument qualification temperature conditions were exceeded in seven of the 10 reactor building control volumes for the STSBO scenario and for one parameter in six of the 10 control volumes. For the LTSBO scenario, instrument qualification temperature conditions for these 15 parameters were exceeded in six of the 10 control volumes. The average temperatures in these six or seven control volumes were hundreds of degrees higher than the instrument qualification temperatures for both STSBO and LTSBO scenarios. The temperature conditions were also exceeded from the time of the transient until the end of the plot, about 40 h for the STSBO and 30 h for the LTSBO.

The instrument qualification conditions for all parameters were for 100% relative humidity, which by definition is a bounding value.

For all 15 plant parameters listed in Table 5, instrument qualification gamma radiation dose conditions were exceeded in nine of the 10 reactor building control volumes for the STSBO scenario, on average by two orders of magnitude. (Note that Table 3 of NUREG/CR-5444 shows the environmental qualifications for radiation dose to be the same for all instruments located in the reactor building.) For the LTSBO scenario, instrument qualification temperature conditions for the 15 parameters were exceeded in all 10 control volumes, typically by a factor of 5×. The gamma dose typically reached the instrument qualification value between approximately 0.5 to 1.5 days from the start of the STSBO and between 5 and 30 days from the start of the LTSBO. There is considerable uncertainty regarding dose rates in the control volumes. The dose rates are based on radionuclide composition within a control volume and do not include cross communications with neighboring control volumes.

The dose rates also do not consider potential shielding factors such as walls, columns, cabinets, etc., or areas where water levels on floors could affect dose rates. Therefore, where dose rates are orders of magnitude higher than qualification values, it could be reasonable to conclude that equipment limits would be exceeded. If dose rates are near qualification values, then a more definitive study may be warranted.

In Table 5, a Category 1 instrument "provides for full qualification, redundancy, and continuous real time display and requires onsite (standby) power" according to NUREG/CR-5444. A Category 2 instrument "provides for qualification, but is less stringent in that it does not (of itself) include seismic qualification, redundancy of continuous display, and requires only a high reliability power source (not necessarily standby power)." Category 1 and 2 instruments have battery backup power as required by NRC Regulatory Guide 1.97, Revision 3, May 1983, [18] when interruption of the instrument is not tolerable.

3.4.4.2 Instruments located in the drywell

Qualification conditions for seven instruments located in the drywell noted in Table 3 of NUREG/CR-5444 [15] are shown in Table 6. These instruments are drywell sump level, drywell radiation, drywell atmosphere temperature, source-range monitors, intermediate-range monitors, average power range monitors, and suppression pool water temperature.

Primary containment control volumes include CV 200 for the lower, CV 201 for the middle, and CV 202 for the upper parts of the drywell and CV 205 for the pedestal area, as shown in Fig. 3. The drywell-to-wetwell vent areas are represented by CV 210, and the wetwell is represented by CV 220. The drywell instruments are assumed to be located in control volumes CV 200 or CV 201. The suppression pool water temperature instrument is assumed to be located in control volume CV 220.

For all seven plant parameters listed in Table 6, instrument qualification pressure conditions in the two relevant drywell control volumes and the wetwell pool control volume were exceeded for STSBO and LTSBO scenarios. Instrument qualification values were exceeded by about 25 psi for the STSBO scenario and by about 35 psi for the LTSBO scenario. The pressure conditions in the other containment control volumes also would have exceeded instrument qualification values typical for the other containment instruments if instrumentation was located there. The qualification conditions were exceeded for about 2 h for the STSBO and 4 h for the LTSBO.

For all seven plant parameters listed in Table 6, instrument qualification temperature conditions were exceeded for the two relevant drywell control volumes (as well as for the remaining two drywell control volumes and the drywell-to-wetwell control volume) for the STSBO and LTSBO scenarios. The temperature in one of the relevant drywell control volumes was more than 100°F higher than the equipment qualification value for the STSBO and LTSBO scenarios. The temperature was about 20°F higher for the other relevant drywell control volume. The temperatures in the other two drywell control volumes were greater than the qualification value by about 2000°F for both scenarios. The drywell-to-wetwell control volume was about 600°F higher for the STSBO scenario and about 10°F higher for the LTSBO scenario. The temperature in the suppression pool control volume was lower than the temperature qualification limit for the STSBO and LTSBO scenarios. The qualification conditions for several of the primary containment control volumes were significantly exceeded for almost the whole duration of the transients plotted for the STSBO and LTSBO, about 40 and 30 h, respectively. The qualification conditions for the remainder of the control volumes were exceeded or approximately equal for almost the whole duration of the transients plotted for the STSBO and LTSBO.

The instrument qualification conditions for all parameters were for 100% relative humidity, which by definition is a bounding value.

For all seven plant parameters listed in Table 6, instrument gamma radiation dose conditions were near instrument qualification values for the relevant drywell control volumes for both the STSBO and LTSBO scenarios. The gamma dose conditions for the drywell-to-wetwell and wetwell control volumes were about 100× and 1000× higher, respectively, than qualification values for the STSBO and LTSBO scenarios. These conditions were reached within hours of the start of the scenario. As noted previously, there is considerable uncertainty regarding dose rates in the control volumes. The dose rates are based on radionuclide composition within a control volume and do not include cross-communications with neighboring control volumes.

The dose rates also do not consider potential shielding factors such as walls, columns, cabinets, etc., or areas where water levels on floors could affect dose rates. Therefore, where dose rates are orders of magnitude higher than qualification values, it would be reasonable to conclude that equipment limits would be exceeded. If dose rates are near qualification values, then more definitive study may be warranted.

3.4.4.3 Instrumentation located on the reactor vessel

A number of thermocouples (46 for one representative BWR reactor vessel) measure vessel temperature to monitor various vessel components to assess vessel stresses during heat-up and cooldown. These are usually Type T thermocouples used for temperatures up to about 589 K (600°F) in this application. Type T thermocouples have a temperature range up to about 623 K (662°F). These thermocouples have potential application for use during accident scenarios to assess various conditions that are reflected by the temperatures at various locations on the reactor vessel walls.

Figures 11 and 12 show reactor pressure vessel outer wall temperatures vs. time estimated for the STSBO and LTSBO scenarios, respectively. (The sequence of readings, COR-TLH.101 – COR-TLH.901, are for locations on the outer wall of the lower head starting from the centerline and moving outward.) A reference line for 623 K (662°F) is provided to indicate where the temperature range of Type T thermocouples is exceeded. It is evident that within hours of the start of the SBO scenarios the vessel wall temperatures significantly exceed the normal Type T thermocouple temperature limit.

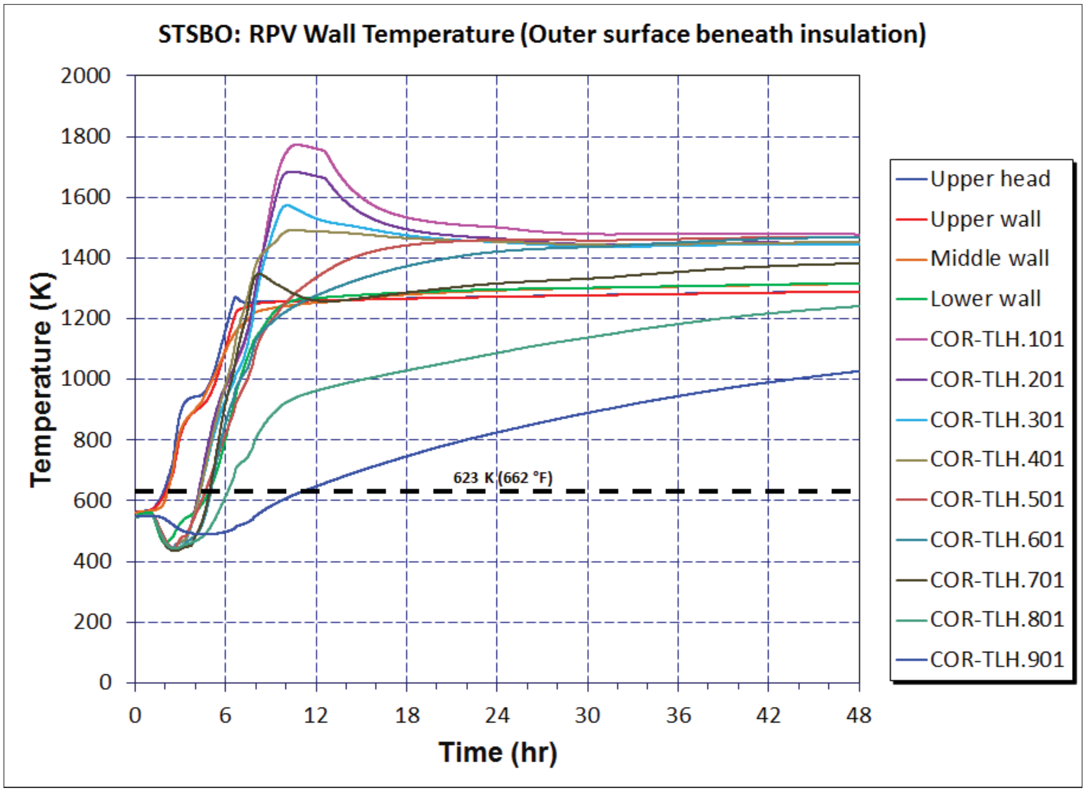


Fig. 13. Reactor vessel wall temperature (STSBO).

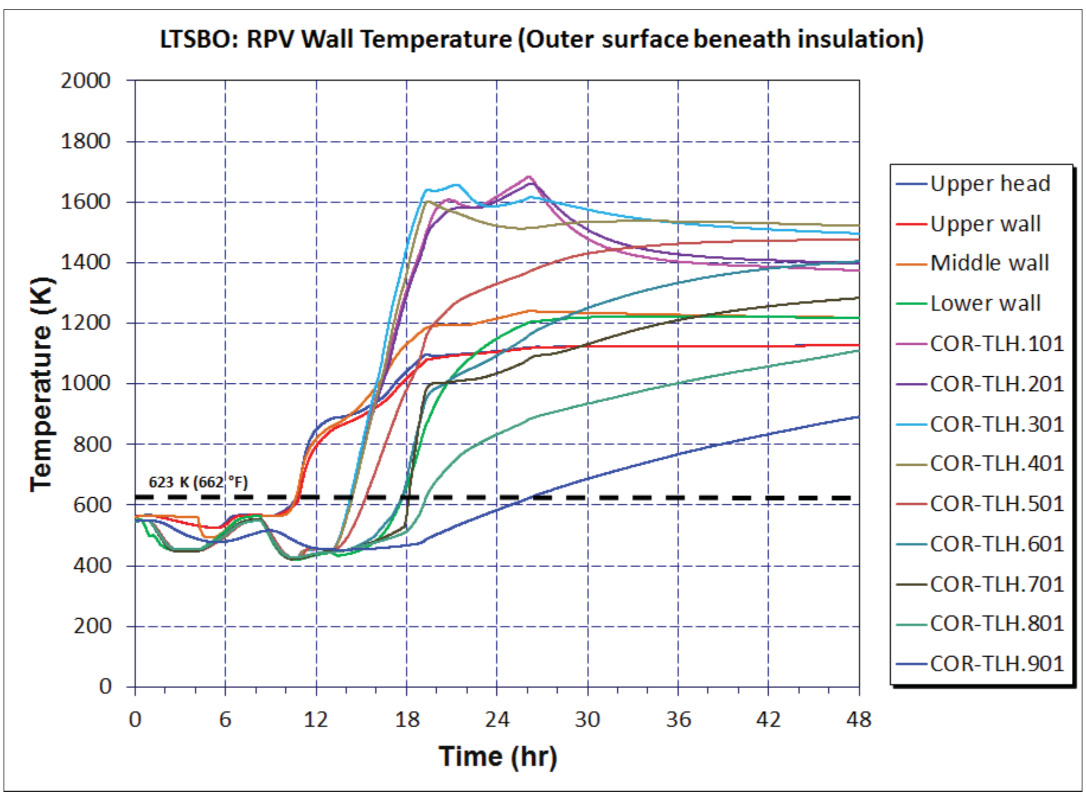


Fig. 14. Reactor vessel wall temperature (LTSBO).

Table 5. Environmental conditions expected for instruments located in the reactor building providing selected key parameters for accident management

Plant parameter	Category	Instrument qualification condition [15]		Reactor building control volume condition exceeds instrument qualification condition for the STSBO scenario (S), LTSBO scenario (L), or both the STSBO and LTSBO scenarios (SL)											
				401	**402**	**403**	**404**	**405**	**406**	**407**	**408**	**409**	**410**	**411**	**412**
Reactor pressure	1	Temperature (°F)	178	SL	SL	SL	S	SL	S			SL		L	SL
		Pressure (psig)	1.5	S											
		Relative humidity (%)	100												
		Gamma radiation dose (Rad)	3.5E+04	SL	SL	SL	SL	SL	SL	SL	SL	SL	SL	L	SL
Reactor vessel water level	1	Temperature (°F)	250	SL	SL	SL		SL	S			SL		L	SL
		Pressure (psig)	0	SL	SL	SL	SL	SL	SL	L	L	L	L	L	L
		Relative humidity (%)	100												
		Gamma radiation dose (Rad)	3.5E+04	SL	SL	SL	SL	SL	SL	SL	SL	SL	SL	L	SL
Suppression pool water level	1	Temperature (°F)	183	SL	SL	SL	S	SL	S			SL		L	SL
		Pressure (psig)	0	SL	SL	SL	SL	SL	SL	L	L	L	L	L	L
		Relative humidity (%)	100												
		Gamma radiation dose (Rad)	3.5E+04	SL	SL	SL	SL	SL	SL	SL	SL	SL	SL	L	SL
Drywell pressure	1	Temperature (°F)	207	SL	SL	SL	S	SL	S			SL		L	SL
		Pressure (psig)	2	S											
		Relative humidity (%)	100												
		Gamma radiation dose (Rad)	3.5E+04	SL	SL	SL	SL	SL	SL	SL	SL	SL	SL	L	SL
Drywell oxygen concentration	1	Temperature (°F)	141	SL	SL	SL	SL	SL	S			SL		SL	SL
		Pressure (psig)	0	SL	SL	SL	SL	SL	SL	L	L	L	L	L	L
		Relative humidity (%)	100												
		Gamma radiation dose (Rad)	3.5E+04	SL	SL	SL	SL	SL	SL	SL	SL	SL	SL	L	SL
Drywell hydrogen concentration	1	Temperature (°F)	141	SL	SL	SL	SL	SL	S			SL		SL	SL
		Pressure (psig)	0	SL	SL	SL	SL	SL	SL	L	L	L	L	L	L
		Relative humidity (%)	100												
		Gamma radiation dose (Rad)	3.5E+04	SL	SL	SL	SL	SL	SL	SL	SL	SL	SL	L	SL

Table 5. Environmental conditions expected for instruments located in the reactor building providing for selected key parameters for accident management (continued)

Plant parameter	Category	Instrument qualification condition [15]		Reactor building control volume condition exceeds instrument qualification condition for the STSBO scenario (S), LTSBO scenario (L), or both the STSBO and LTSBO scenarios (SL)											
				401	402	403	404	405	406	407	408	409	410	411	412
Suppression pool spray flow	2	Temperature (°F)	150	SL	SL	SL	SL	SL	S			SL		SL	SL
		Pressure (psig)	0	SL	SL	SL	SL	SL	SL	L	L	L	L	L	L
		Relative humidity (%)	100												
		Gamma radiation dose (Rad)	3.5E+04	SL	SL	SL	SL	SL	SL	SL	SL	SL	SL	L	SL
Drywell spray flow	2	Temperature (°F)	150	SL	SL	SL	SL	SL	S			SL		SL	SL
		Pressure (psig)	1	S											
		Relative humidity (%)	100												
		Gamma radiation dose (Rad)	3.5E+04	SL	SL	SL	SL	SL	SL	SL	SL	SL	SL	L	SL
Reactor core isolation cooling system flow	2	Temperature (°F)	120	SL	SL	SL	SL	SL	S			SL	S	SL	SL
		Pressure (psig)	1	S											
		Relative humidity (%)	100												
		Gamma radiation dose (Rad)	3.5E+04	SL	SL	SL	SL	SL	SL	SL	SL	SL	SL	L	SL
High-pressure coolant injection system flow	2	Temperature (°F)	120	SL	SL	SL	SL	SL	S			SL	S	SL	SL
		Pressure (psig)	2	S											
		Relative humidity (%)	100												
		Gamma radiation dose (Rad)	3.5E+04	SL	SL	SL	SL	SL	SL	SL	SL	SL	SL	L	SL
Core spray system flow	2	Temperature (°F)	120	SL	SL	SL	SL	SL	S			SL	S	SL	SL
		Pressure (psig)	2	S											
		Relative humidity (%)	100												
		Gamma radiation dose (Rad)	3.5E+04	SL	SL	SL	SL	SL	SL	SL	SL	SL	SL	L	SL
Low-pressure coolant injection system flow	2	Temperature (°F)	120	SL	SL	SL	SL	SL	S			SL	S	SL	SL
		Pressure (psig)	2	S											
		Relative humidity (%)	100												
		Gamma radiation dose (Rad)	3.5E+04	SL	SL	SL	SL	SL	SL	SL	SL	SL	SL	L	SL

Table 5. Environmental conditions expected for instruments located in the reactor building providing for selected key parameters for accident management (continued)

Plant parameter	Category	Instrument qualification condition [15]		Reactor building control volume condition exceeds instrument qualification condition for the STSBO scenario (S), LTSBO scenario (L), or both the STSBO and LTSBO scenarios (SL)											
				401	**402**	**403**	**404**	**405**	**406**	**407**	**408**	**409**	**410**	**411**	**412**
RHR system flow	2	Temperature (°F)	120	SL	SL	SL	SL	SL	S			SL	S	SL	SL
		Pressure (psig)	2	S											
		Relative humidity (%)	100												
		Gamma radiation dose (Rad)	3.5E+04	SL	SL	SL	SL	SL	SL	SL	SL	SL	SL	L	SL
Reactor core isolation cooling system room temperature	2	Temperature (°F)	120	SL	SL	SL	SL	SL	S			SL	S	SL	SL
		Pressure (psig)	2	S											
		Relative humidity (%)	100												
		Gamma radiation dose (Rad)	3.5E+04	SL	SL	SL	SL	SL	SL	SL	SL	SL	SL	L	SL
High-pressure coolant injection system room temperature	2	Temperature (°F)	120	SL	SL	SL	SL	SL	S			SL	S	SL	SL
		Pressure (psig)	2	S											
		Relative humidity (%)	100												
		Gamma radiation dose (Rad)	3.5E+04	SL	SL	SL	SL	SL	SL	SL	SL	SL	SL	L	SL

Table 6. Environmental conditions expected for instruments located in the drywell or wetwell providing selected key parameters for accident management

Plant parameter	Category	Instrument qualification condition [15]		Reactor building control volume condition exceeds instrument qualification condition for the STSBO scenario (S), LTSBO scenario (L), or both the STSBO and LTSBO scenarios (SL)					
				200	**201**	**202**	**205**	**210**	**220**
Drywell sump level	1	Temperature (°F)	317	SL	SL	SL	SL	SL	
		Pressure (psig)	49	SL	SL	SL	SL	SL	SL
		Relative humidity (%)	100						
		Gamma radiation dose (Rad)	4.4E+07	S			S	SL	SL
Drywell atmosphere temperature	2	Temperature (°F)	317	SL	SL	SL	SL	SL	
		Pressure (psig)	49	SL	SL	SL	SL	SL	SL
		Relative humidity (%)	100						
		Gamma radiation dose (Rad)	4.4E+07	S			S	SL	SL
Drywell radiation	1	Temperature (°F)	317	SL	SL	SL	SL	SL	
		Pressure (psig)	49	SL	SL	SL	SL	SL	SL
		Relative humidity (%)	100						
		Gamma radiation dose (Rad)	4.4E+07	S			S	SL	SL
Source-range monitors	1	Temperature (°F)	317	SL	SL	SL	SL	SL	
		Pressure (psig)	49	SL	SL	SL	SL	SL	SL
		Relative humidity (%)	100						
		Gamma radiation dose (Rad)	4.4E+07	S			S	SL	SL
Intermediate-range monitors	1	Temperature (°F)	317	SL	SL	SL	SL	SL	
		Pressure (psig)	49	SL	SL	SL	SL	SL	SL
		Relative humidity (%)	100						
		Gamma radiation dose (Rad)	4.4E+07	S			S	SL	SL
Average power range monitors	1	Temperature (°F)	317	SL	SL	SL	SL	SL	
		Pressure (psig)	49	SL	SL	SL	SL	SL	SL
		Relative humidity (%)	100						
		Gamma radiation dose (Rad)	4.4E+07	S			S	SL	SL
Suppression pool water temperature (torus shell)	1	Temperature (°F)	317	SL	SL	SL	SL	SL	
		Pressure (psig)	49		SL	SL	SL	SL	SL
		Relative humidity (%)	100						
		Gamma radiation dose (Rad)	3.5E+07	S			S	SL	SL

4. CONCLUSION

The objective of this research is to estimate the environmental conditions that essential instrumentation must survive to remain functional following risk-dominant severe accidents at a BWR/4-Mark I reactors to improve severe accident management capabilities. This could lead to improvements in instrumentation survivability and performance under challenging operating conditions—a major reason for this research program—for BWR/4-Mark I and potentially to other BWR designs.

Loss of electric power and harsh, but as yet unquantified, environmental conditions were experienced at the Fukushima Daiichi units that affected instrumentation performance. Instrumentation failed or provided inaccurate readings or misleading or inconsistent trends, thus complicating operator responses and accident management.

The MELCOR severe accident code was used to examine postulated unmitigated STSBO and LTSBO severe accident scenarios for a BWR/4 with a Mark I containment developed as part of the SOARCA project sponsored by the US NRC to improve reactor accident analysis efforts. These analysis results were used as a basis for estimating environmental conditions in the reactor primary containment and reactor building during these scenarios. Instrumentation that provides essential measurement parameters and values necessary for informing accident management and mitigation efforts is subject to these environmental conditions.

This study found that instrumentation for about 20 reactor parameters deemed critical to informing operator responses and accident management activities would have exceeded qualification values for the STSBO and LTSBO severe accident scenarios. Instrumentation located in the drywell would typically have exceeded qualification values for pressure and temperature. Instrumentation located in the wetwell would have exceeded qualification values for pressure and radiation dose. Instrumentation located in the reactor building would typically have exceeded qualification values for temperature and radiation dose. Instrument qualification values would typically have been exceeded in the early hours and days of the accident scenarios during the period of core damage, hydrogen generation, core relocation from the reactor vessel to the drywell, and breach of containment. It is not presumed, nor is it true, that the fact that an instrument exceeds its qualification conditions fails given margin incumbent in the designs and specifications.

As observed in the Fukushima Daiichi accident [1], some instrumentation did fail and other instruments responded erratically, inconsistently, inaccurately, and indicated opposite trends. It would be a reasonable conclusion to use instrumentation suspected to have exceeded its qualification conditions with suspicion given that margins may be uncertain, and performance near margins is uncertain. Until forensics examinations of the Fukushima Daiichi instrumentation is performed, the effects on all instrument components, sensors, cables, electronics, indicators is not known, nor is the magnitude of these effects compared to effects on sensing lines or reference legs.

The conclusion of this study is that given unmitigated STSBO or LTSBO accident scenarios at a reactor of BWR/4-Mark I design, the performance of critical instrumentation would be considered suspect because environmental qualification values were exceeded. Accident responses, such as through emergency procedures or severe accident management guidelines, that rely on instrumentation for critical parameters may be affected by potential widespread instrument failure or inaccurate and/or misleading instrument indications during these severe accident scenarios.

This conclusion, however, is tempered by several issues, including (1) uncertainties associated with the MELCOR code to estimate equipment environmental conditions, and (2) unavailability of forensics analysis of instrumentation at the Fukushima Daiichi stations at this time and the related subject of how instrumentation currently in use performs in harsh conditions.

The MELCOR code was used in this study to estimate potential severe accident instrument environmental conditions. However, current MELCOR models were not designed to provide environmental conditions within the primary containment and reactor building with a high degree of specificity and precision; this likely affects temperature conditions more so than pressure. It cannot accommodate, for example, conditions at specific equipment locations or environmental effects from nearby structures or components (i.e., protection, shielding, high-temperature components, etc.). MELCOR itself does not calculate radiation dose, but an algorithm was developed for estimating time-dependent radiation dose in a node based on the fission product inventory in the node. Limitations are that it does not include (1) the contribution to dose from neighboring nodes, including debris on the containment floor; (2) potential effects of a steam environment (that is, some shielding compared to a dry environment) and structures within the nodes that could provide shielding to certain components, depending on their specific locations within a node; or (3) ability to account for radiation from particles deposited on or near specific components.

Forensics analyses of the Fukushima Daiichi instrumentations will be performed in the future. Results of these analyses would have been helpful in understanding the results of the environmental conditions predicted in the MELCOR analyses and in assessing the value of the margin in the instrument designs and on the relative effects of sensors, cables, transmitters, electronics, etc., as well as sensing lines and reference legs.

The objective of this study, to estimate the environmental conditions that essential instrumentation must survive to remain functional following risk-dominant severe accidents, leads to reasonable follow-up questions for consideration given the uncertainty about instrument performance under harsh conditions. How long will particular parameters need to be monitored? How long and under what conditions will specific instruments dependably indicate the values of these parameters? How will operators know when the instruments degrade? And when they degrade, will they have predictable behavior?

Future work may address improvements in the MELCOR models to provide better certainty in estimates of environmental conditions, especially radiation effects. There will also be improvements gained based on results of Fukushima Daiichi forensics analyses. Better knowledge of instrument performance under harsh conditions could then lead to better instruments, alternative measurement technologies, and improvements in accident management strategies and plans.

5. REFERENCES

[1] D. Clayton and W. Poore, *Fukushima Daiichi - A Case Study for BWR Instrumentation and Control Systems Performance During a Severe Accident-Revision 1*, ORNL/TM-2013/154, UT-Battelle, LLC, Oak Ridge National Laboratory, May 2014.

[2] D. Clayton and W. Poore, *DOE-NE Letter Report: Key Parameters for Operator Diagnosis of BWR Plant Condition During a Severe Accident*, ORNL/LTR-2014/320, UT-Battelle, LLC, Oak Ridge National Laboratory, September 2014.

[3] [Online]. Available: http://melcor.sandia.gov.

[4] Sandia Natinoal Laboratories, *State-of-the-Art Reactor Consequence Analyses Project, Volume 1: Peach Bottom Integrated Analysis*, NUREG/CR-7110, U.S. Nuclear Regulatory Commission, January 2012.

[5] R. Chang et al., *State-of-the-Art Reactor Consequence Analyses (SOARCA) Report*, NUREG-1935, Parts 1 and 2, U.S. Nuclear Regulatory Commission, November 2012.

[6] Tokyo Electric Power Company, Inc., *The Paramters Related to the Plants in Fukushima Daiichi Nuclear Power Station*, Archives 2011. [Online]. Available: www.tepco.co.jp/en/nu/fukushima-np/f1/pla/2011/index-e.html.

[7] J. Rempe and D. Knudson, *TMI-2 - A Case Study for PWR Instrumentation Performance During a Severe Accident*, INL/EXT-13-28043 Revision 1, May 2014.

[8] *10 C.F.R. § 50.34(f).*

[9] US Nuclear Regulatory Commission, *Recommendations for Enhancing Reactor Safety in the 21st Century - The Near-Term Task Force Review of Insights from the Fukushima Daiichi Accident*, July 2011. [Online]. Available: pbadupws.nrc.gov/docs/ML1125/ML112510271.pdf..

[10] *10 C.F.R. § 50.44.*

[11] *10 C.F.R. § 50.62.*

[12] *10 C.F.R. § 50.63.*

[13] *10 C.F.R. § 50 Appendix R.*

[14] S. Greene, *The Canary, the Ostrich, and the Black Swan: An Historical Perspective on our Understanding of BWR Severe Accidents and their Mitigation,* San Diego, CA: American Nuclear Society, 2012.

[15] W. Arcieri and D. Hanson, *Instrumentation Availability During Severe Accidents for a Boiling Water Reactor with the Mark 1 Containment*, NUREG/CR-5444, Idaho National Laboratory, February 1992.

[16] *Investigation Committee on the Accident at the Fukushima Nuclear Power Stations of Tokyo Electric Power Company, Interim Report*, 26 December 2011. [Online]. Available: www.cas.go.jp/jp/seisaku/icanps/eng/interim-report.html .

[17] *10 C.F.R. § 50, Appendix A.*

[18] U.S. Nuclear Regulatory Commission, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Environs Conditions During and Following an Accident," in *Regulatory Guide 1.97, Revision 3*, May 1983.

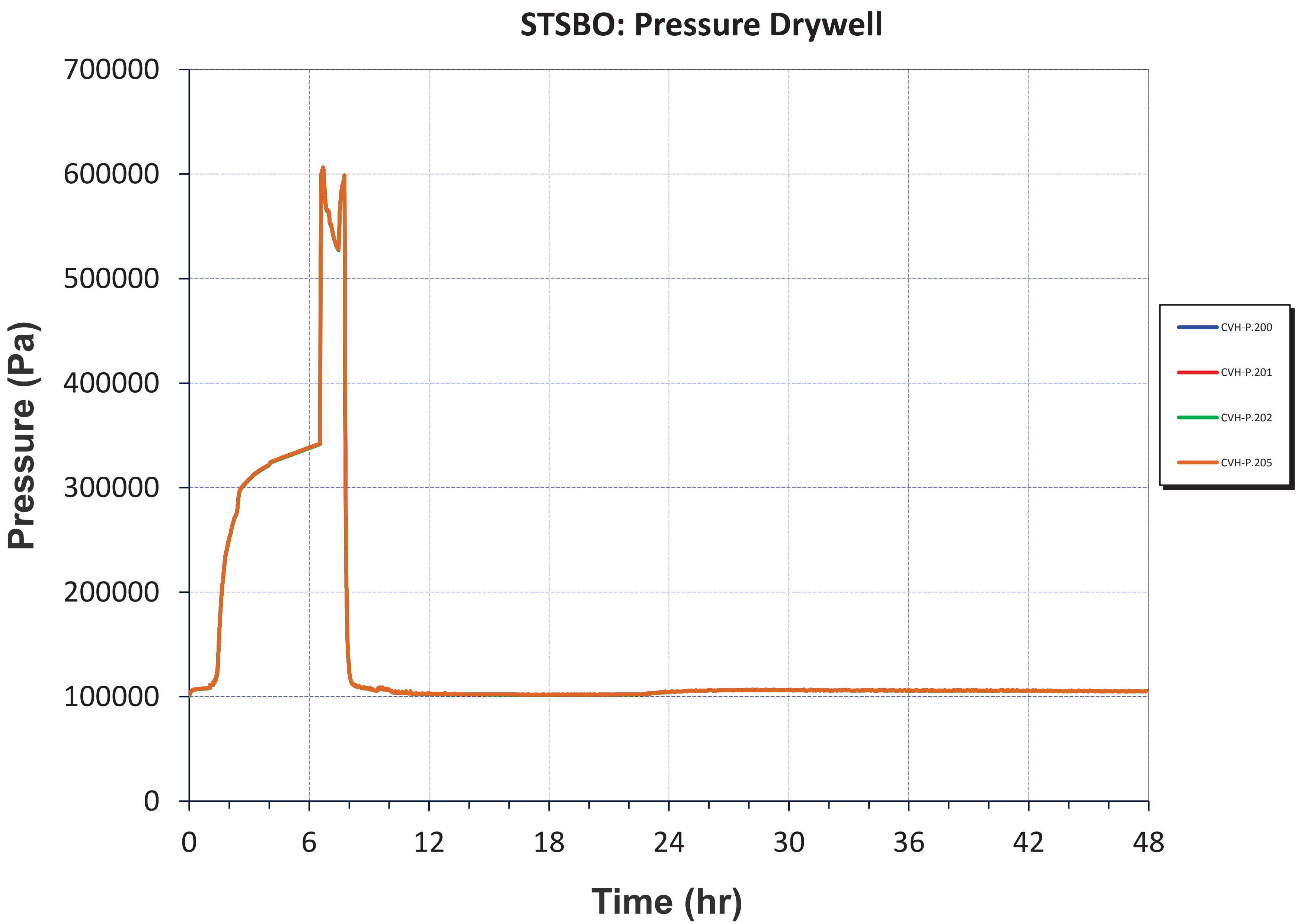
[19] U.S. Nuclear Regulatory Commission , "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," in *Regulatory Guide 1.97, Revision 4*, June 2006.

[20] "IEEE publications may be purchased from the IEEE Service Center," [Online]. Available:

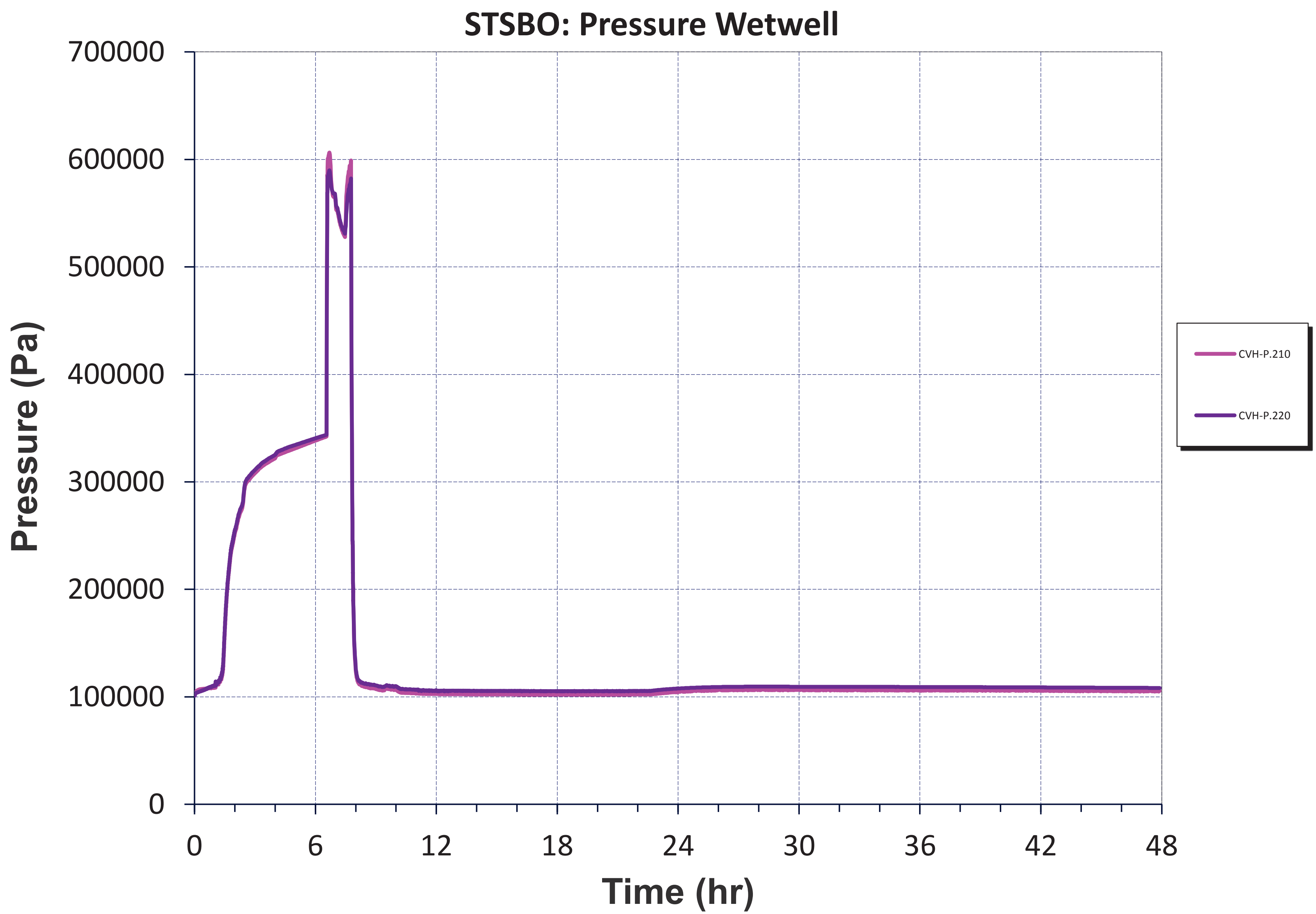
http://www.ieee.org.

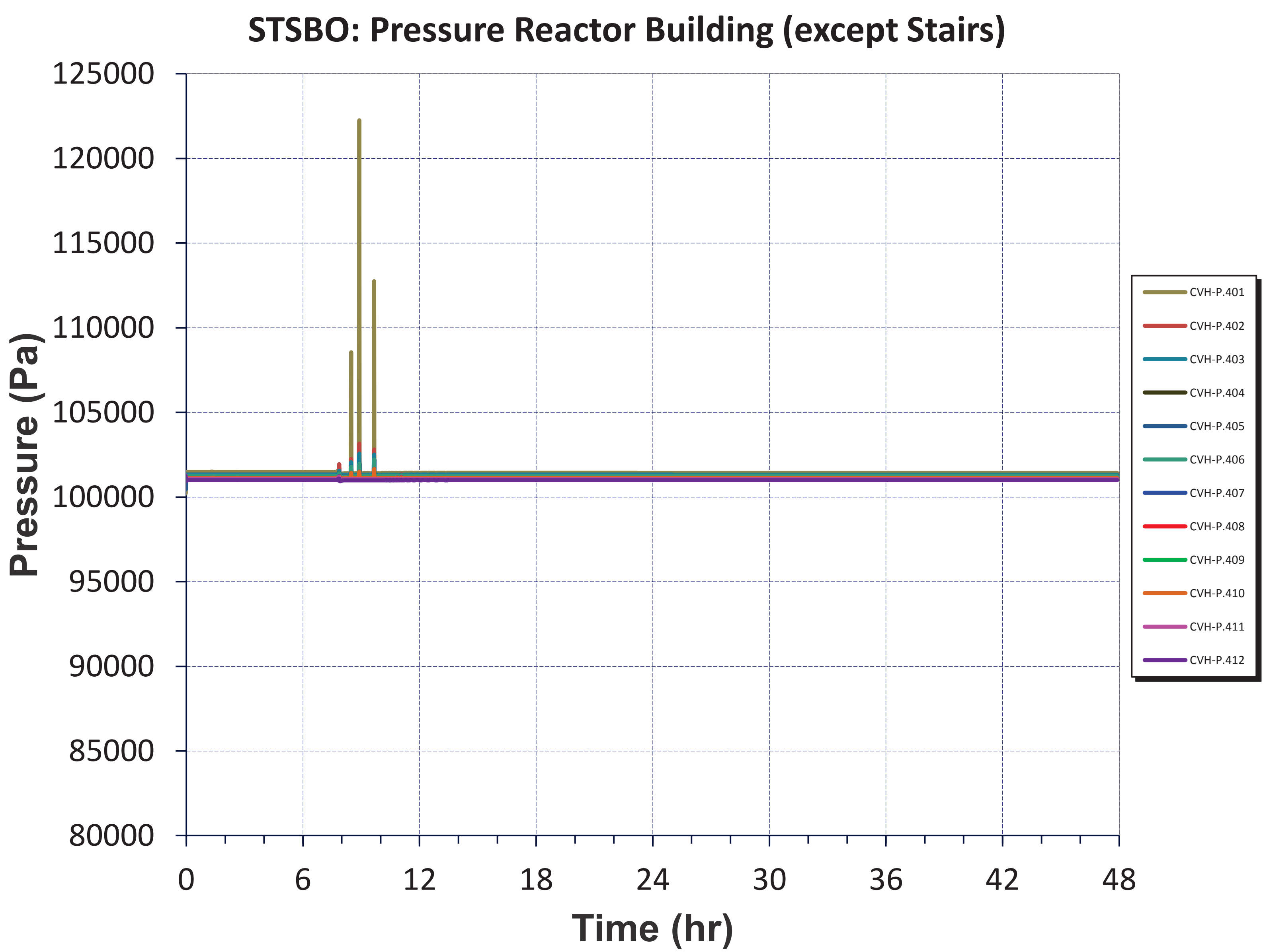
[21] U.S. NRC Order Ea-12-051, Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation, March 2012.

APPENDIX A: STSBO

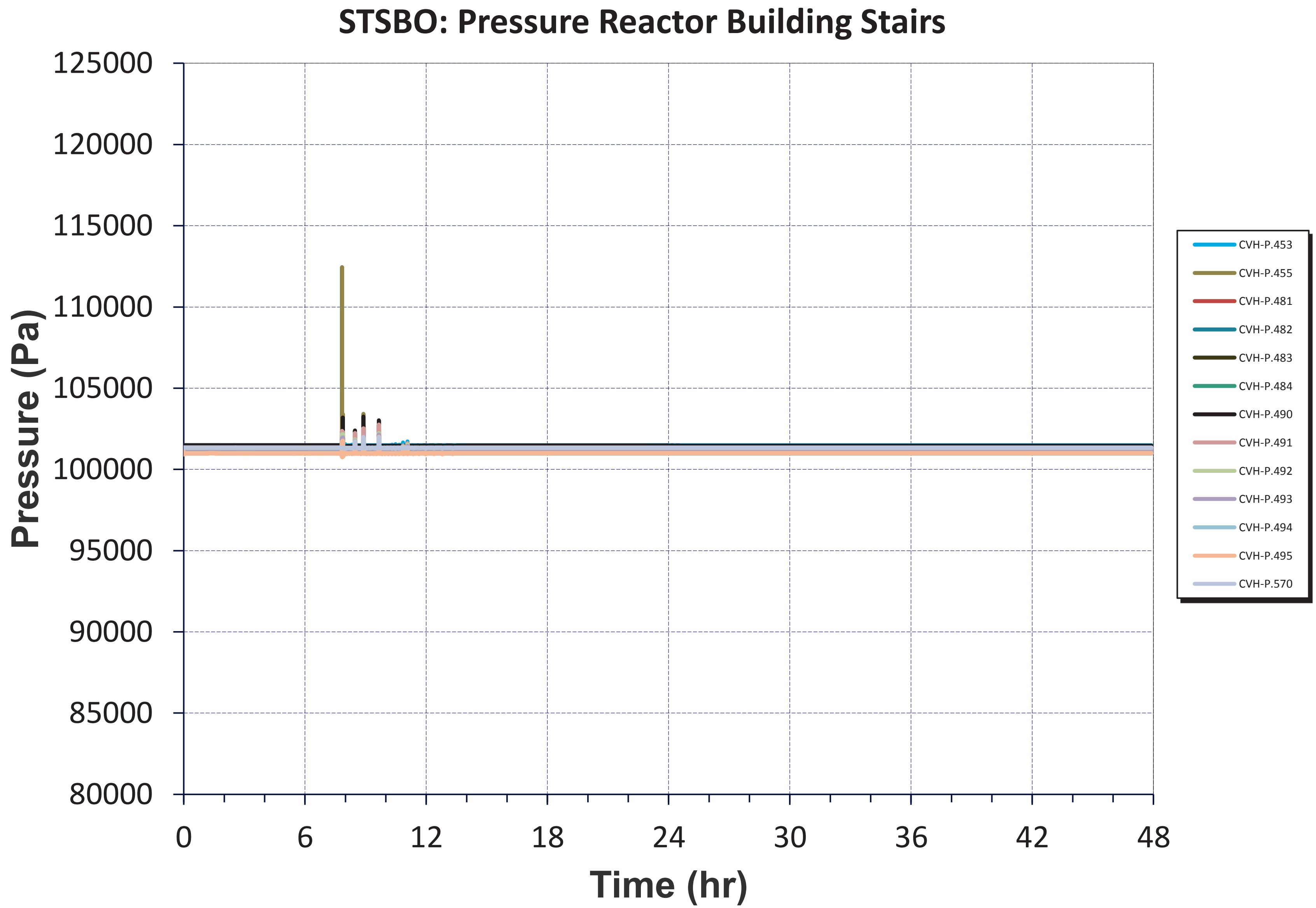


STSBO: Pressure Drywell
Pressure (Pa)
700000
600000
500000
400000
300000
200000
100000
0
0
6
12
18
24
30
36
42
48
Time (hr)
CVH-P.200
CVH-P.201
CVH-P.202
CVH-P.205

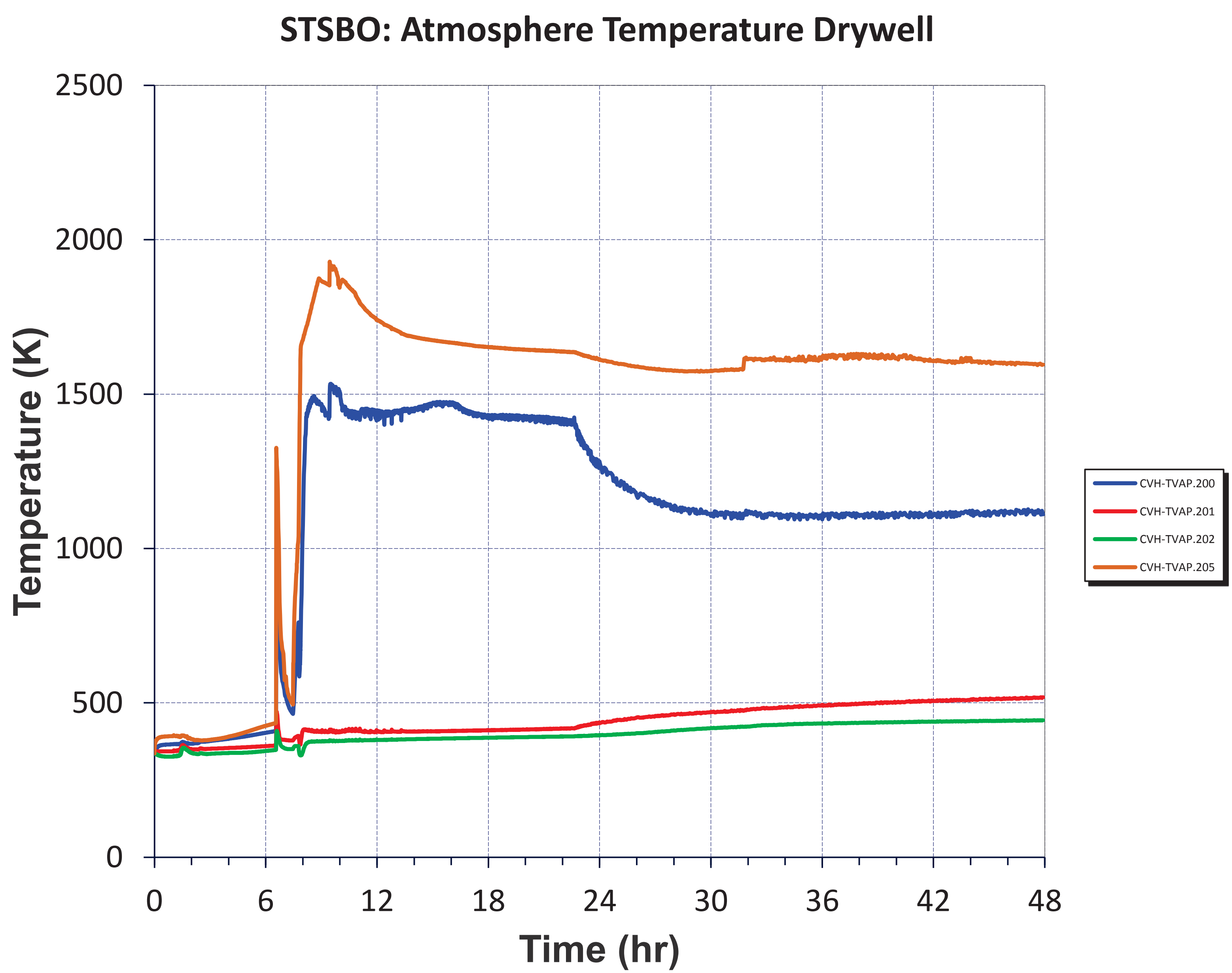

STSBO: Pressure Wetwell
Pressure (Pa)
700000
600000
500000
400000
300000
200000
100000
0
0
6
12
18
24
30
36
42
48
Time (hr)
CVH-P.210
CVH-P.220



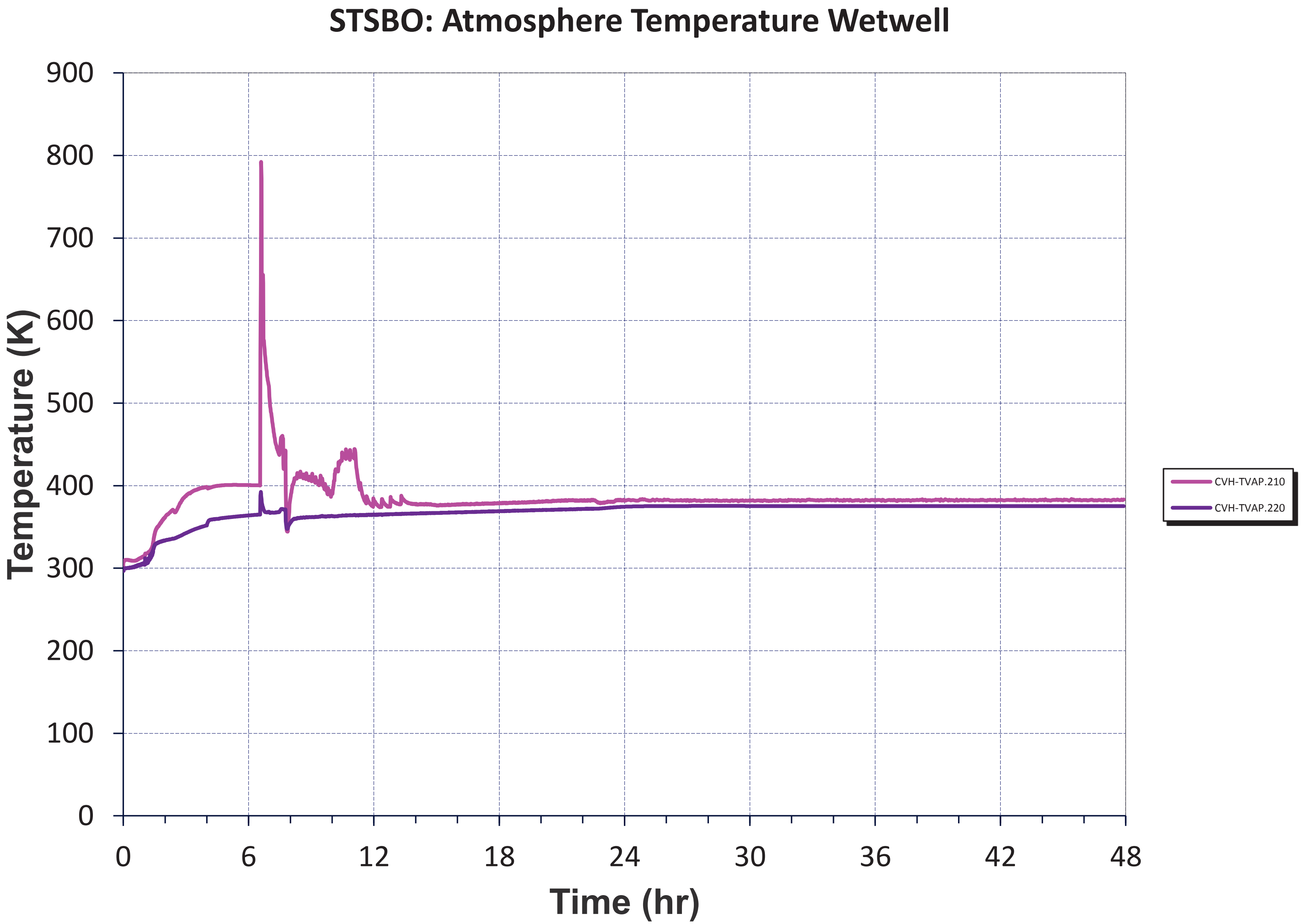
STSBO: Pressure Reactor Building (except Stairs)
Pressure (Pa)
125000
120000
115000
110000
105000
100000
95000
90000
85000
80000
0
6
12
18
24
30
36
42
48
Time (hr)
CVH-P.401
CVH-P.402
CVH-P.403
CVH-P.404
CVH-P.405
CVH-P.406
CVH-P.407
CVH-P.408
CVH-P.409
CVH-P.410
CVH-P.411
CVH-P.412



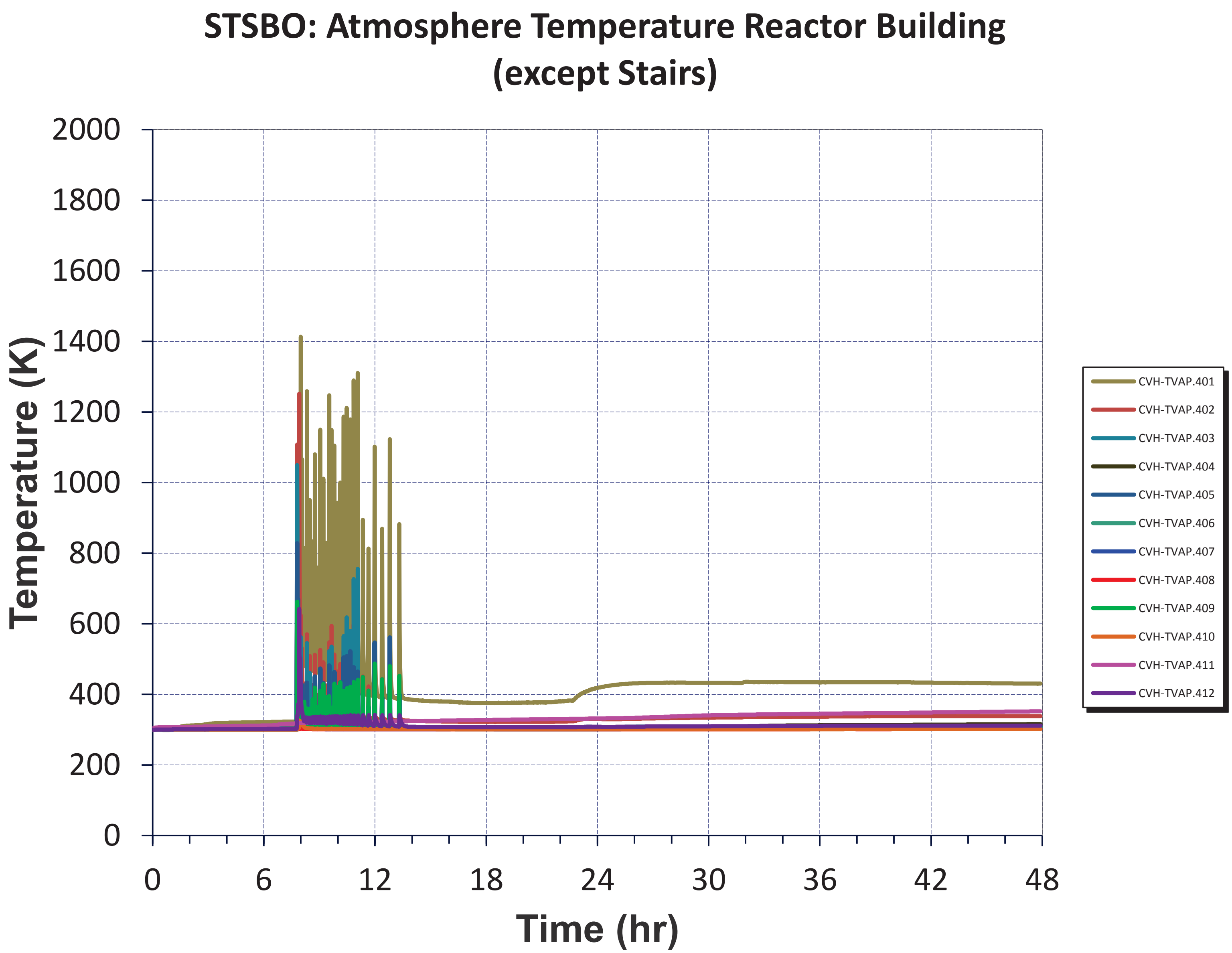
STSBO: Pressure Reactor Building Stairs
Pressure (Pa)
125000
120000
115000
110000
105000
100000
95000
90000
85000
80000
0
6
12
18
24
30
36
42
48
Time (hr)
CVH-P.453
CVH-P.455
CVH-P.481
CVH-P.482
CVH-P.483
CVH-P.484
CVH-P.490
CVH-P.491
CVH-P.492
CVH-P.493
CVH-P.494
CVH-P.495
CVH-P.570

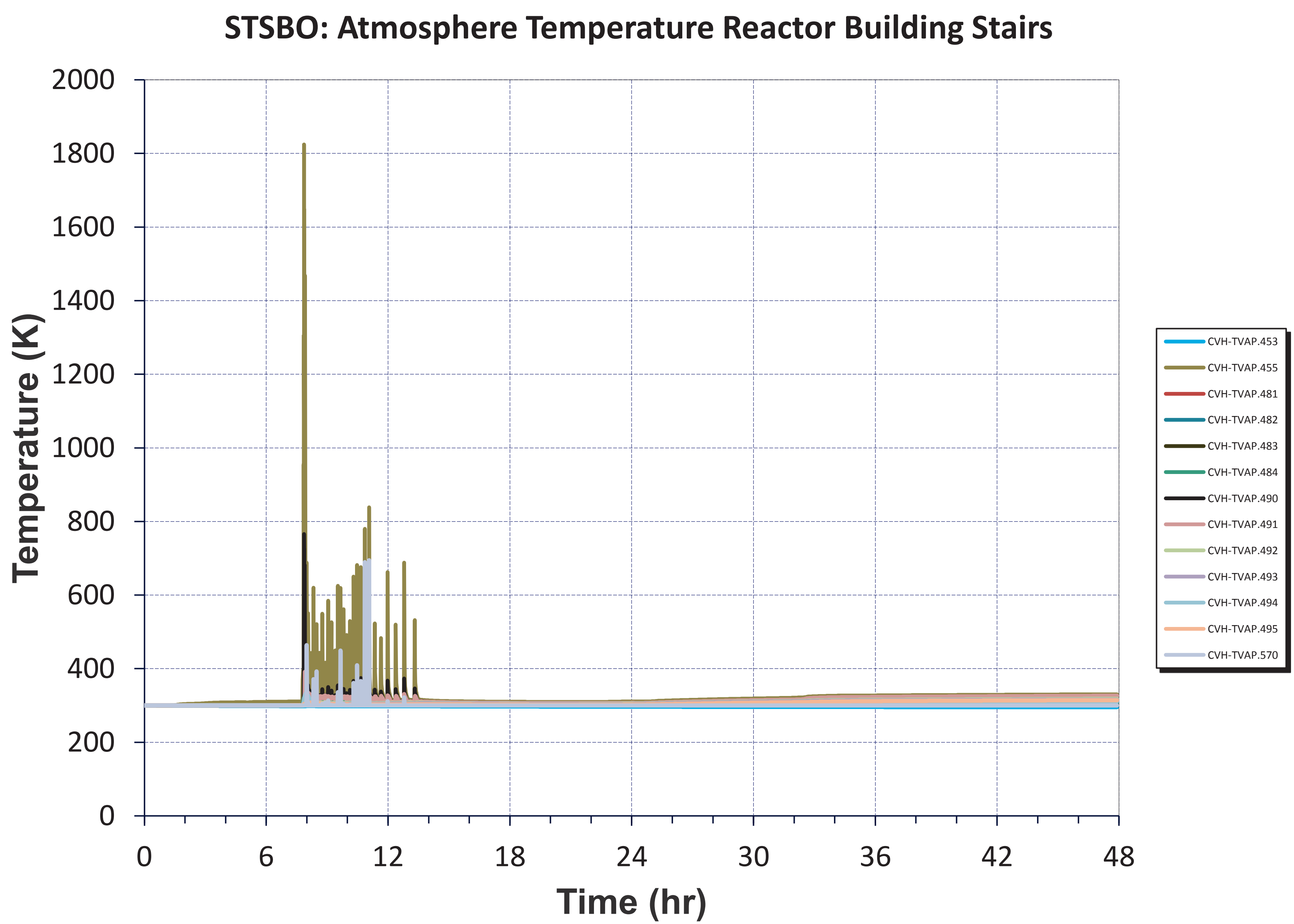


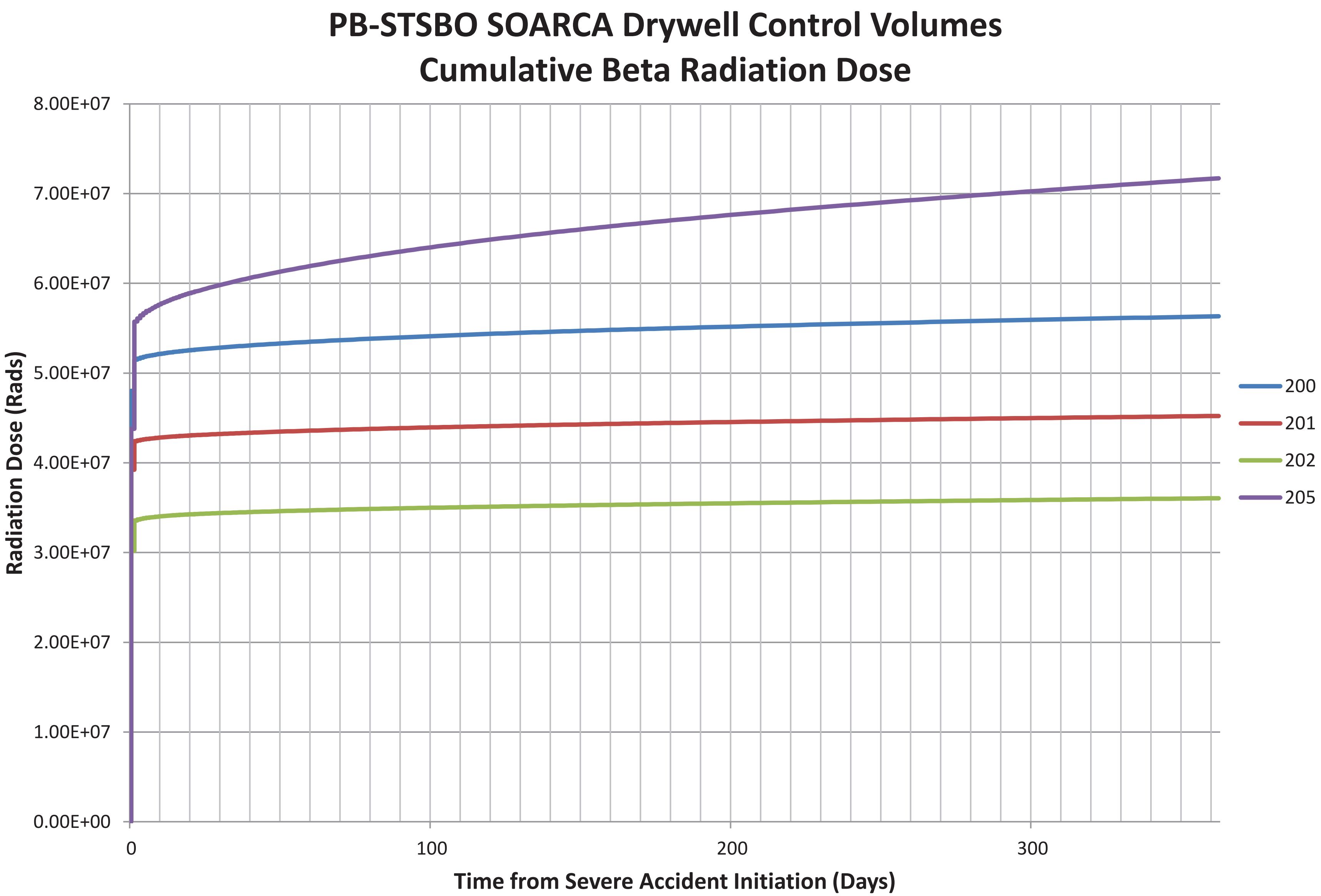
STSBO: Atmosphere Temperature Drywell
Temperature (K)
2500
2000
1500
1000
500
0
0
6
12
18
24
30
36
42
48
Time (hr)
CVH-TVAP.200
CVH-TVAP.201
CVH-TVAP.202
CVH-TVAP.205

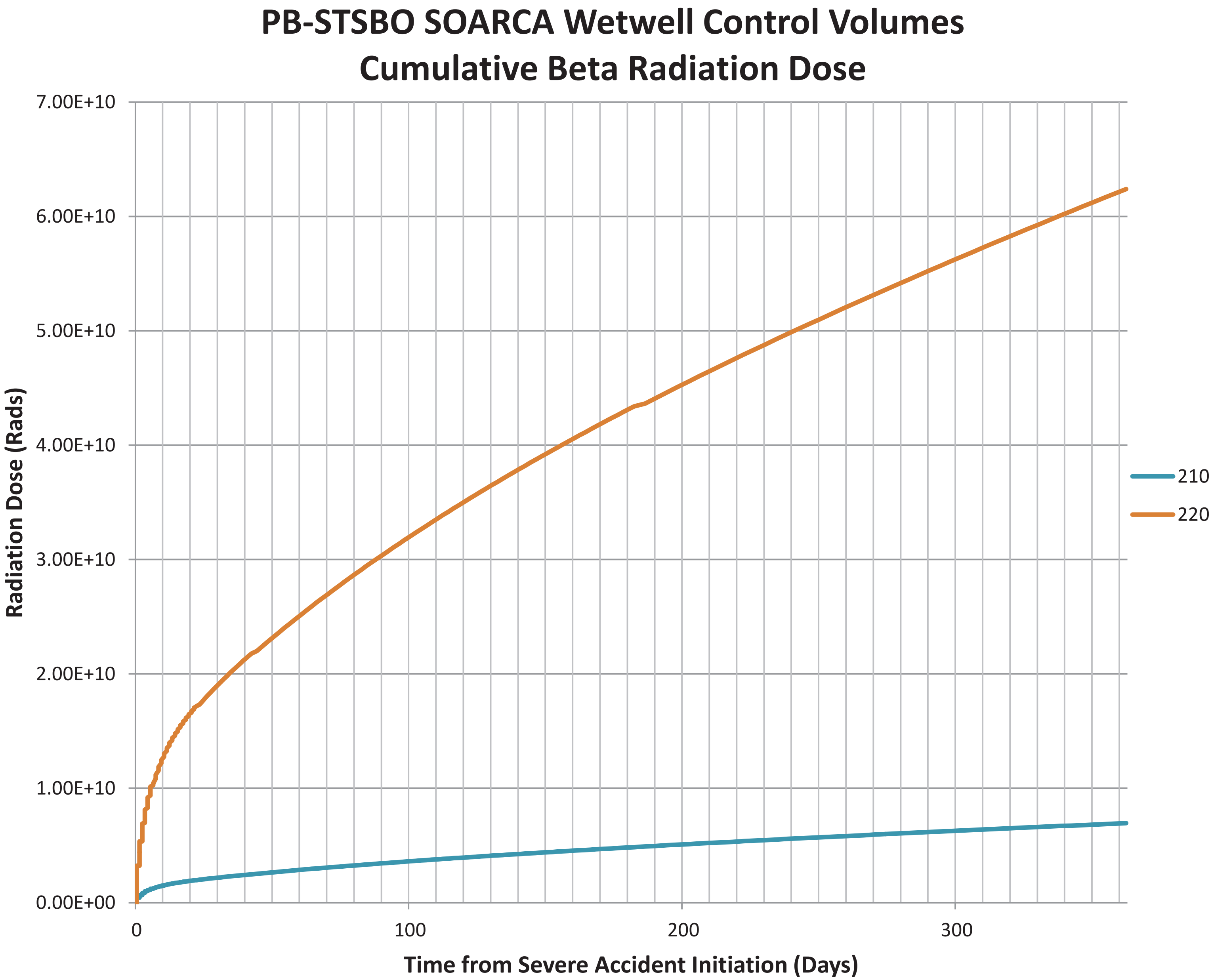


STSBO: Atmosphere Temperature Wetwell
Temperature (K)
900
800
700
600
500
400
300
200
100
0
0
6
12
18
24
30
36
42
48
Time (hr)
CVH-TVAP.210
CVH-TVAP.220

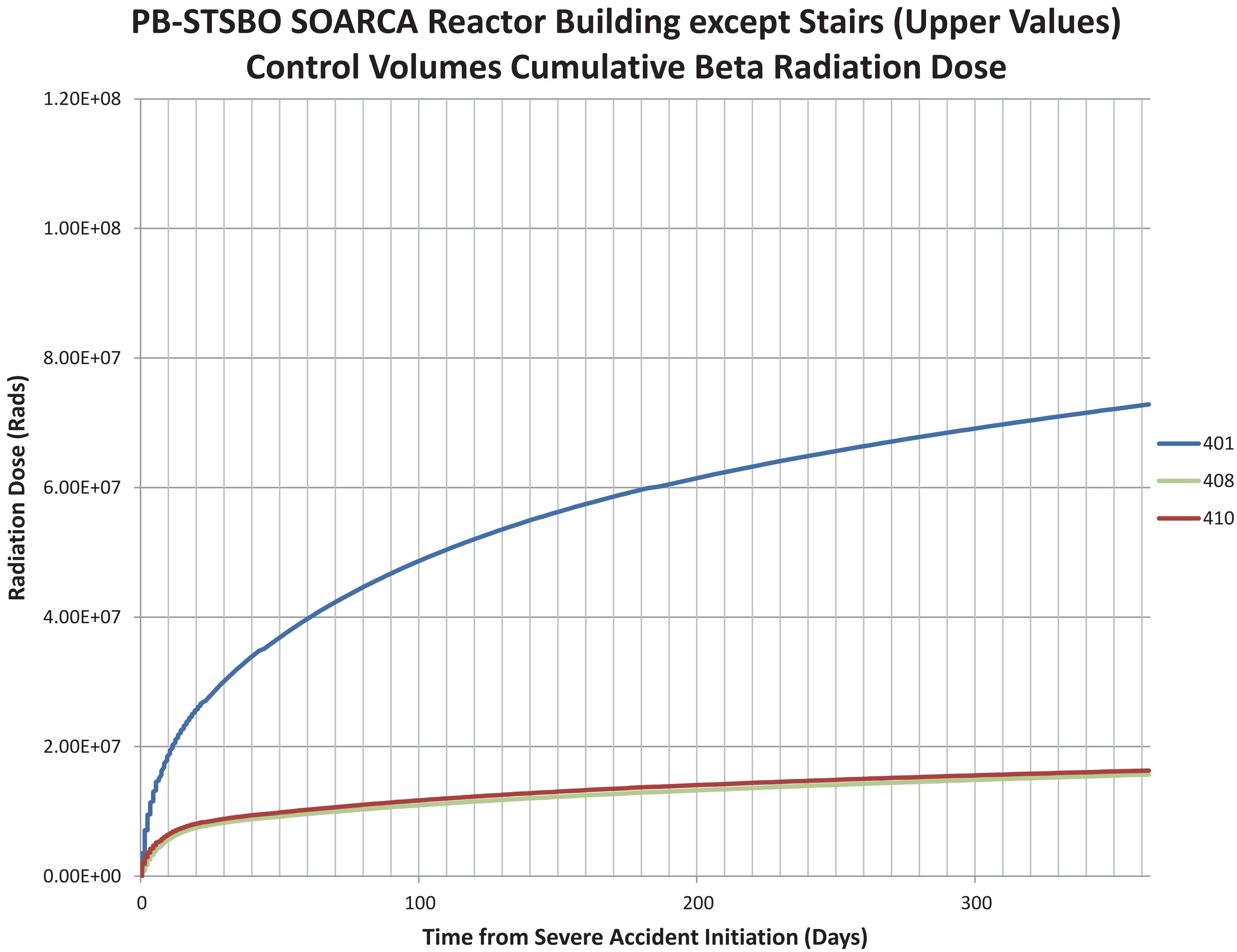

STSBO: Atmosphere Temperature Reactor Building
(except Stairs)
Temperature (K)
Time (hr)
0
200
400
600
800
1000
1200
1400
1600
1800
2000
0
6
12
18
24
30
36
42
48
CVH-TVAP.401
CVH-TVAP.402
CVH-TVAP.403
CVH-TVAP.404
CVH-TVAP.405
CVH-TVAP.406
CVH-TVAP.407
CVH-TVAP.408
CVH-TVAP.409
CVH-TVAP.410
CVH-TVAP.411
CVH-TVAP.412

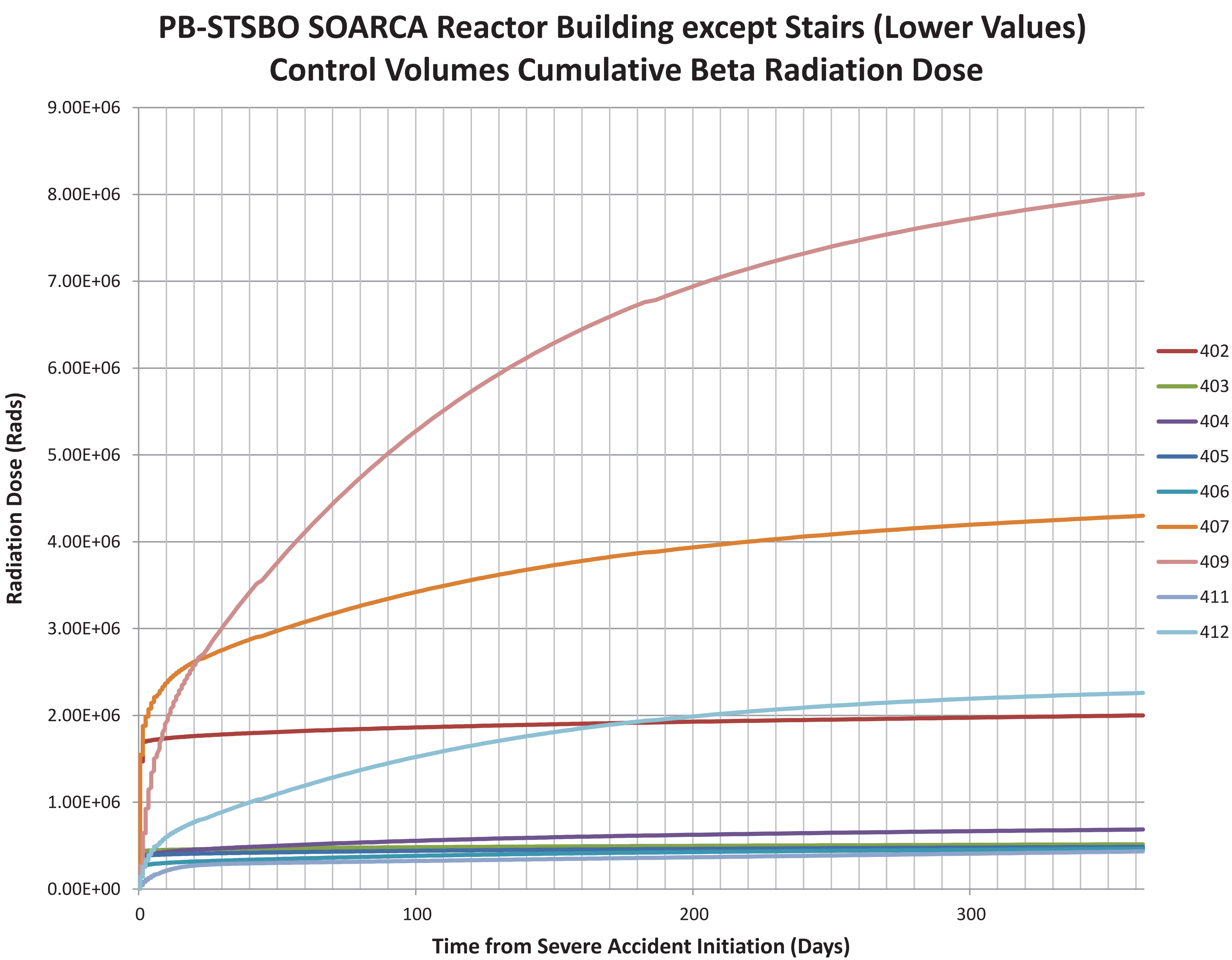

STSBO: Atmosphere Temperature Reactor Building Stairs
Temperature (K)
2000
1800
1600
1400
1200
1000
800
600
400
200
0
0
6
12
18
24
30
36
42
48
Time (hr)
CVH-TVAP.453
CVH-TVAP.455
CVH-TVAP.481
CVH-TVAP.482
CVH-TVAP.483
CVH-TVAP.484
CVH-TVAP.490
CVH-TVAP.491
CVH-TVAP.492
CVH-TVAP.493
CVH-TVAP.494
CVH-TVAP.495
CVH-TVAP.570

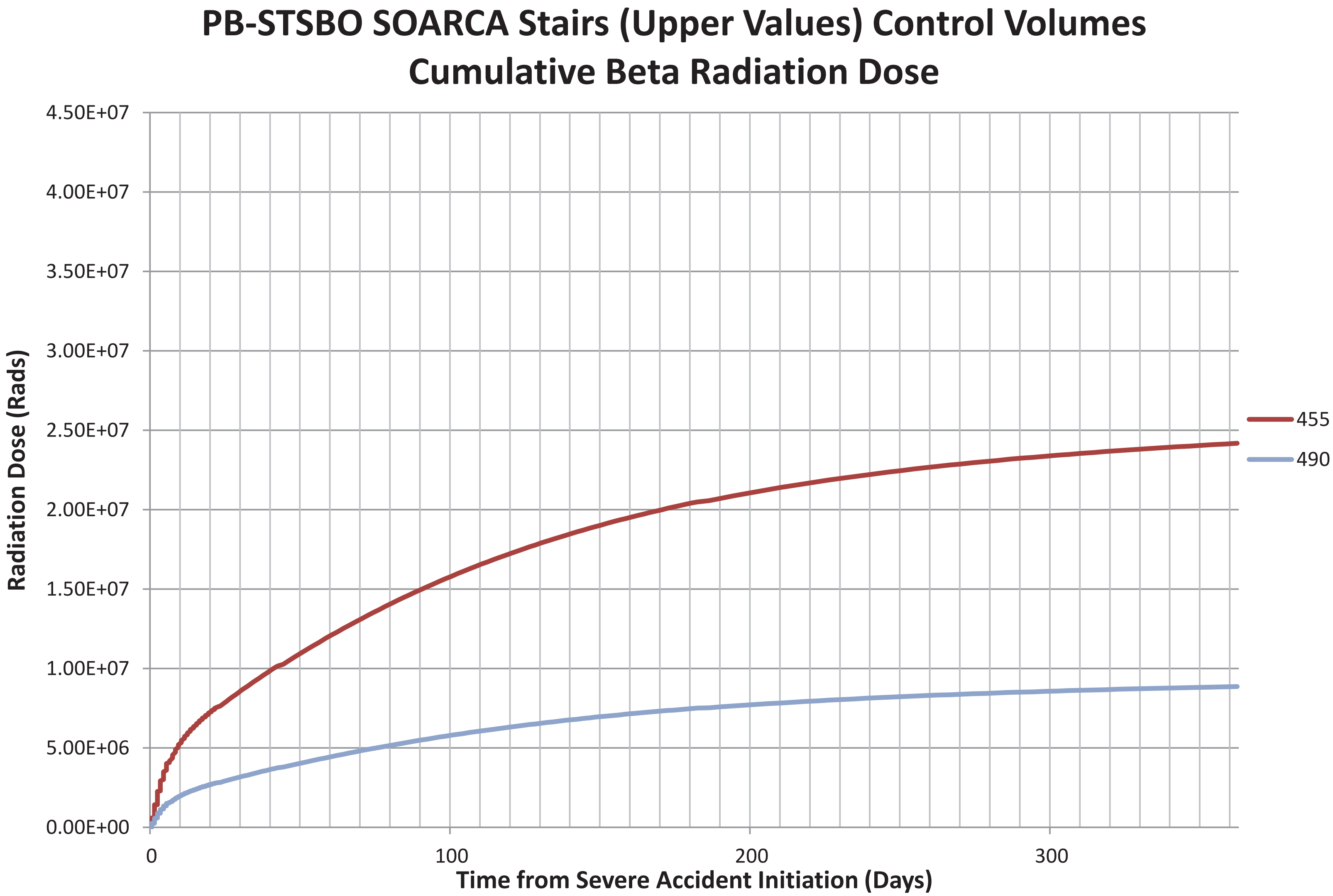

PB-STSBO SOARCA Drywell Control Volumes
Cumulative Beta Radiation Dose
8.00E+07
7.00E+07
6.00E+07
5.00E+07
4.00E+07
3.00E+07
2.00E+07
1.00E+07
0.00E+00
Radiation Dose (Rads)
0
100
200
300
Time from Severe Accident Initiation (Days)
200
201
202
205

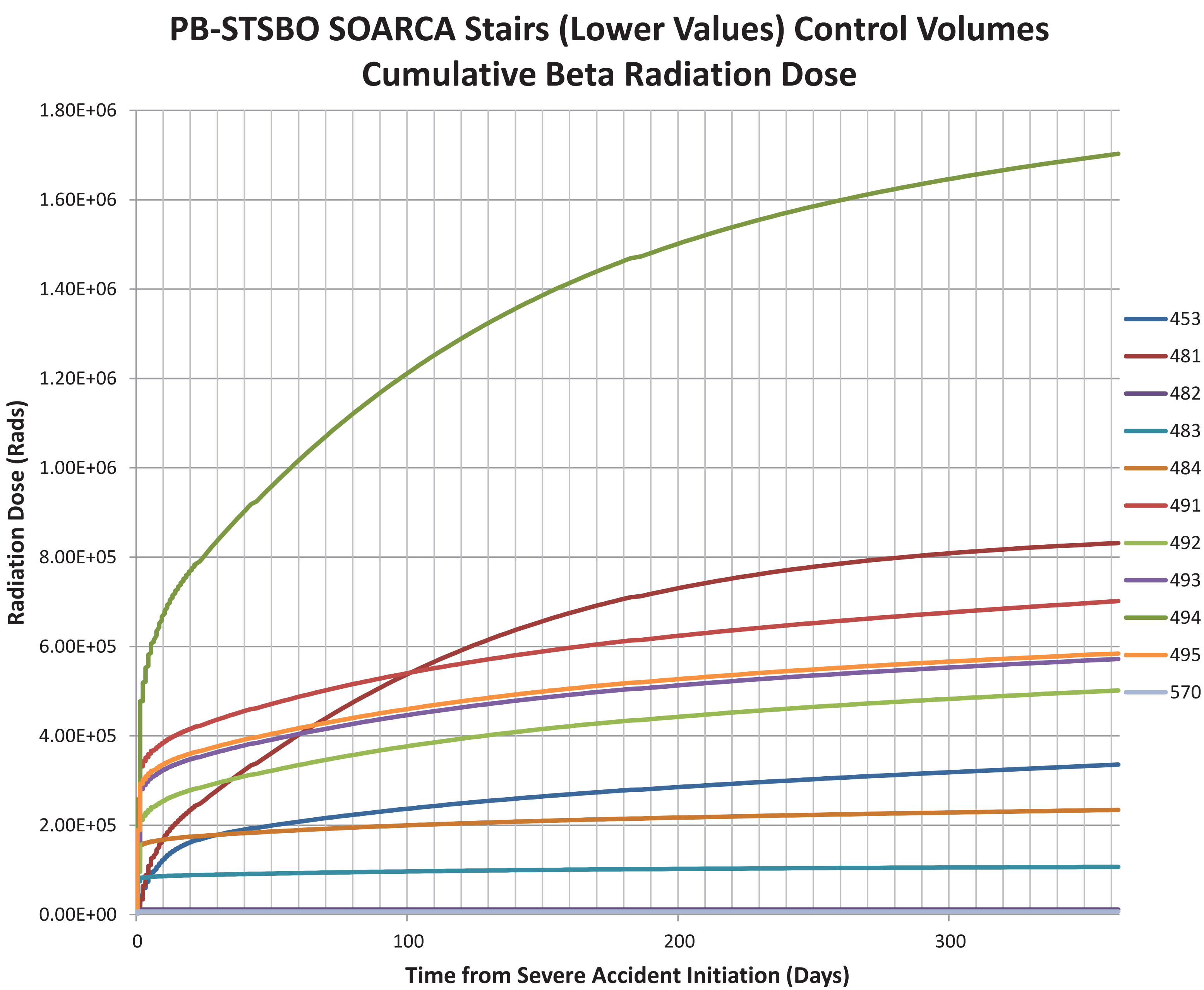


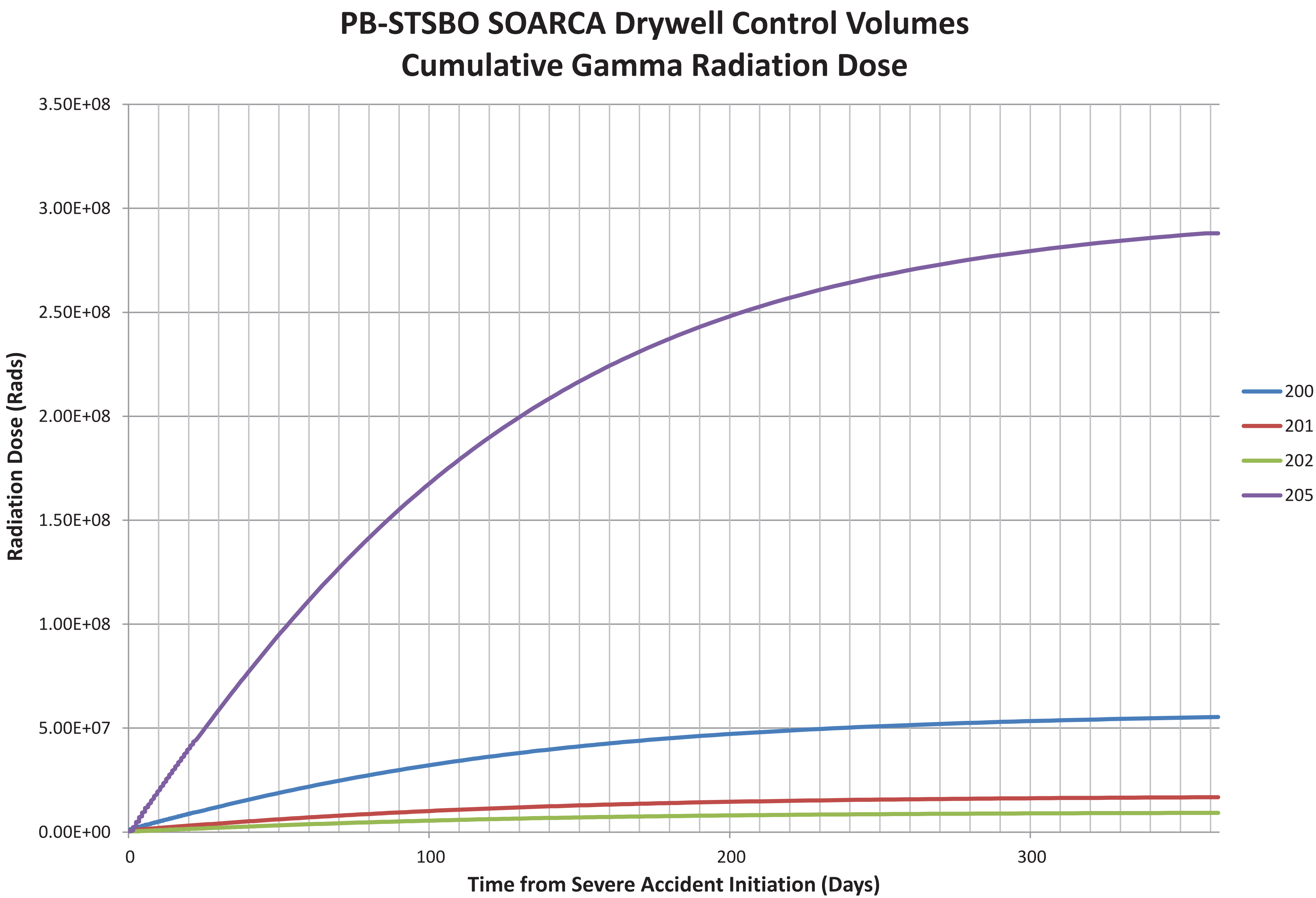
PB-STSBO SOARCA Wetwell Control Volumes
Cumulative Beta Radiation Dose
7.00E+10
6.00E+10
5.00E+10
4.00E+10
3.00E+10
2.00E+10
1.00E+10
0.00E+00
Radiation Dose (Rads)
0
100
200
300
Time from Severe Accident Initiation (Days)
210
220


PB-STSBO SOARCA Reactor Building except Stairs (Upper Values)
Control Volumes Cumulative Beta Radiation Dose
Radiation Dose (Rads)
1.20E+08
1.00E+08
8.00E+07
6.00E+07
4.00E+07
2.00E+07
0.00E+00
0
100
200
300
Time from Severe Accident Initiation (Days)
401
408
410

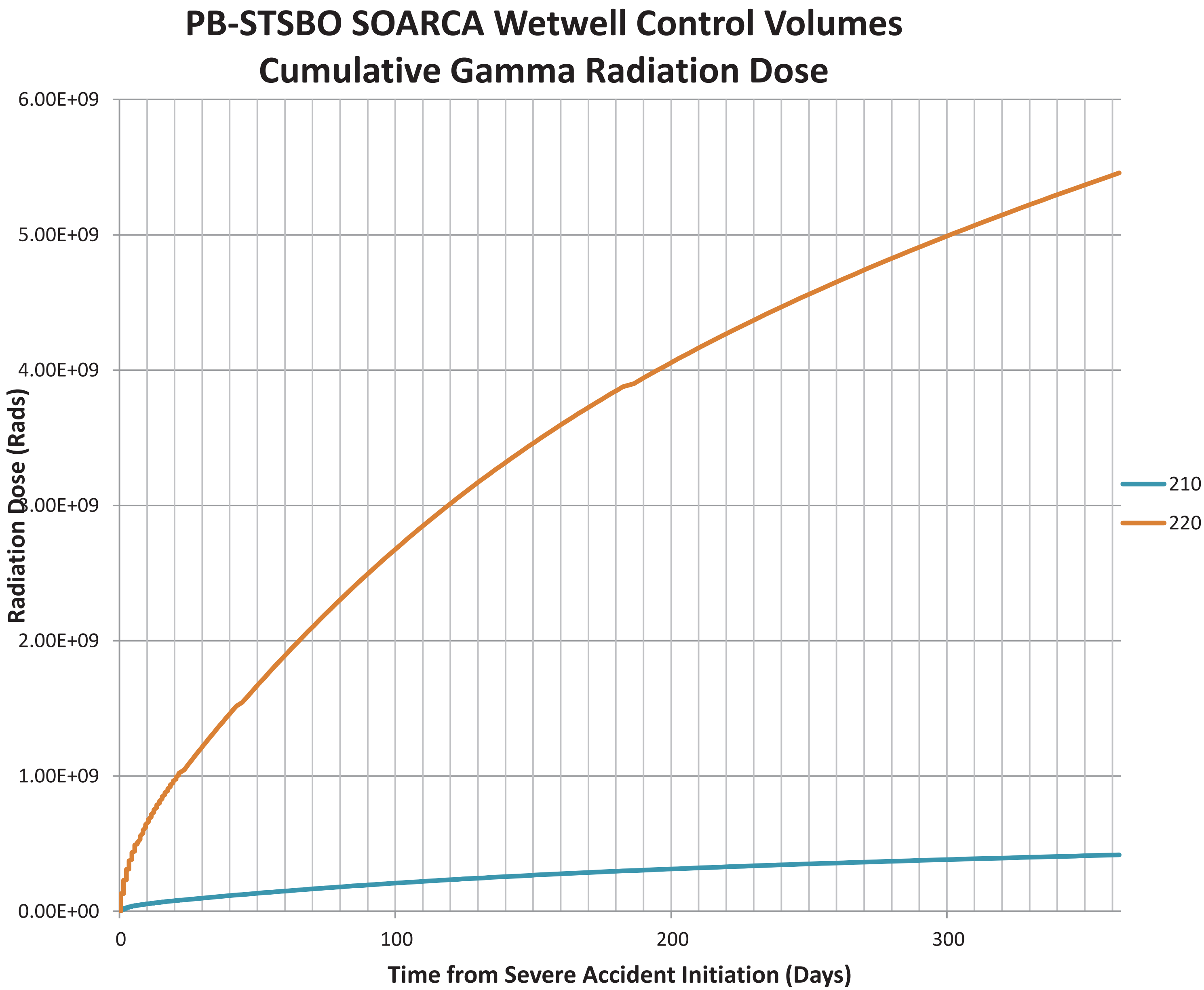

PB-STSBO SOARCA Reactor Building except Stairs (Lower Values)
Control Volumes Cumulative Beta Radiation Dose
Radiation Dose (Rads)
9.00E+06
8.00E+06
7.00E+06
6.00E+06
5.00E+06
4.00E+06
3.00E+06
2.00E+06
1.00E+06
0.00E+00
0
100
200
300
Time from Severe Accident Initiation (Days)
402
403
404
405
406
407
409
411
412

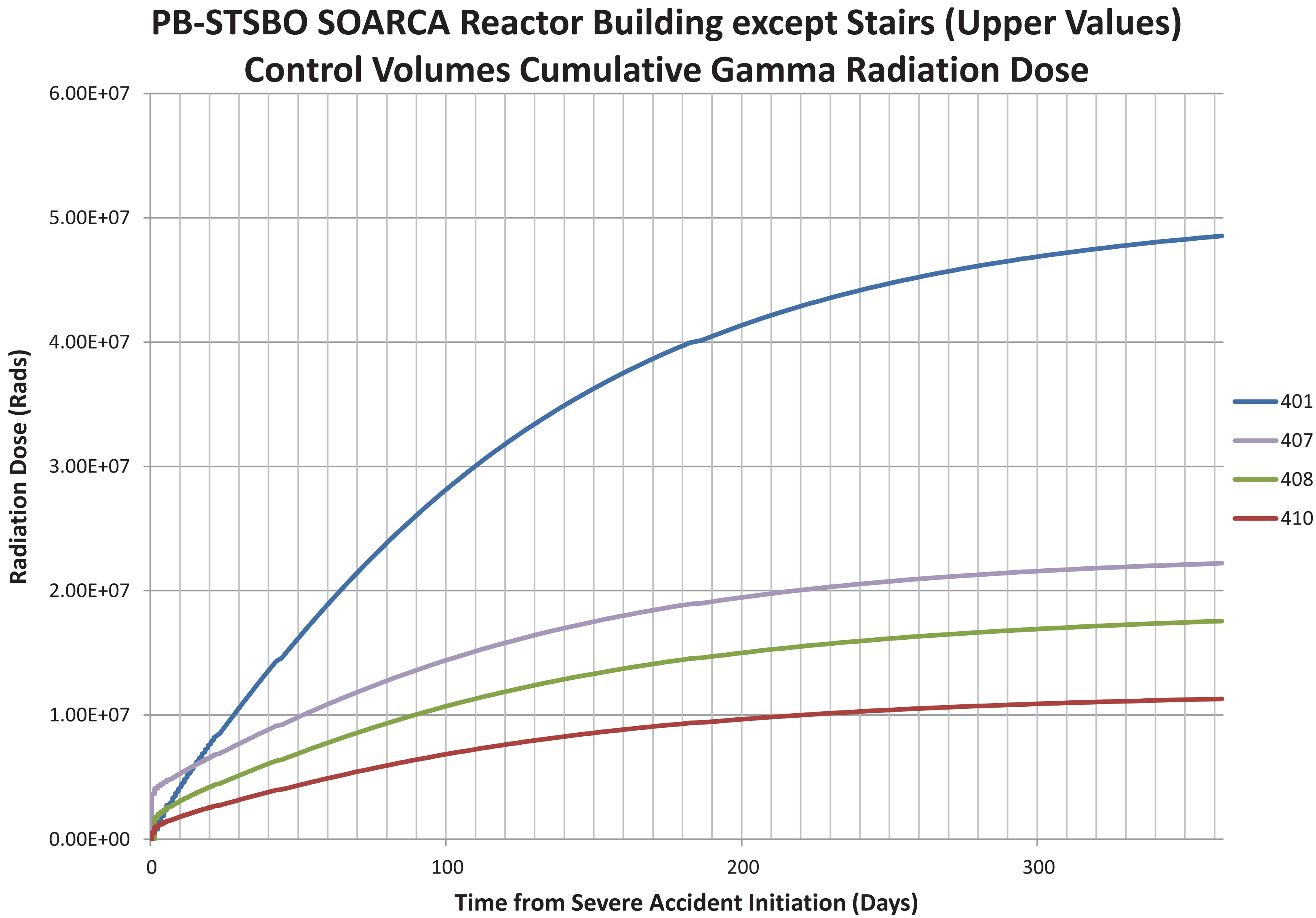

PB-STSBO SOARCA Stairs (Upper Values) Control Volumes
Cumulative Beta Radiation Dose
Radiation Dose (Rads)
4.50E+07
4.00E+07
3.50E+07
3.00E+07
2.50E+07
2.00E+07
1.50E+07
1.00E+07
5.00E+06
0.00E+00
0
100
200
300
Time from Severe Accident Initiation (Days)
455
490


PB-STSBO SOARCA Stairs (Lower Values) Control Volumes
Cumulative Beta Radiation Dose
Radiation Dose (Rads)
1.80E+06
1.60E+06
1.40E+06
1.20E+06
1.00E+06
8.00E+05
6.00E+05
4.00E+05
2.00E+05
0.00E+00
0
100
200
300
Time from Severe Accident Initiation (Days)
453
481
482
483
484
491
492
493
494
495
570

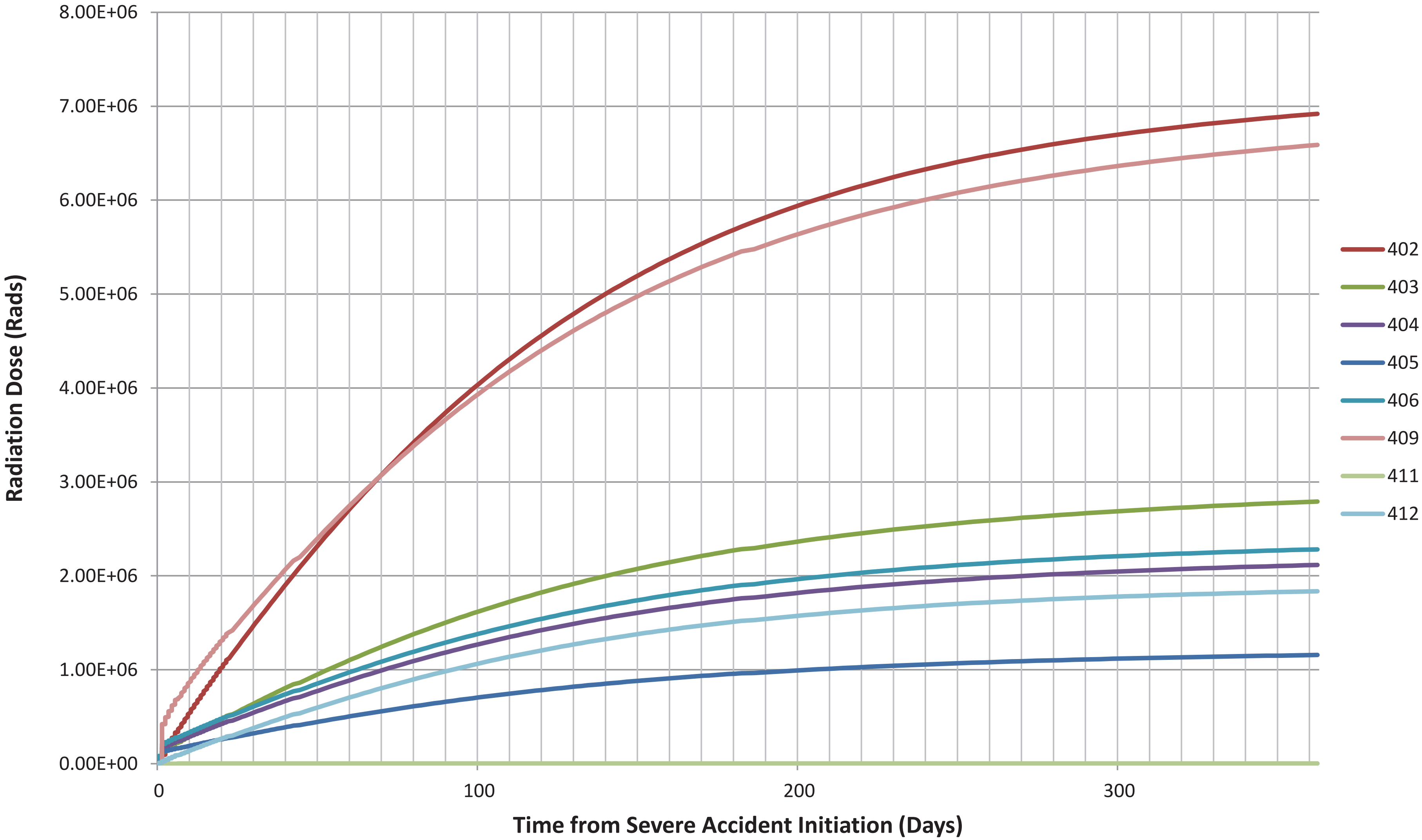


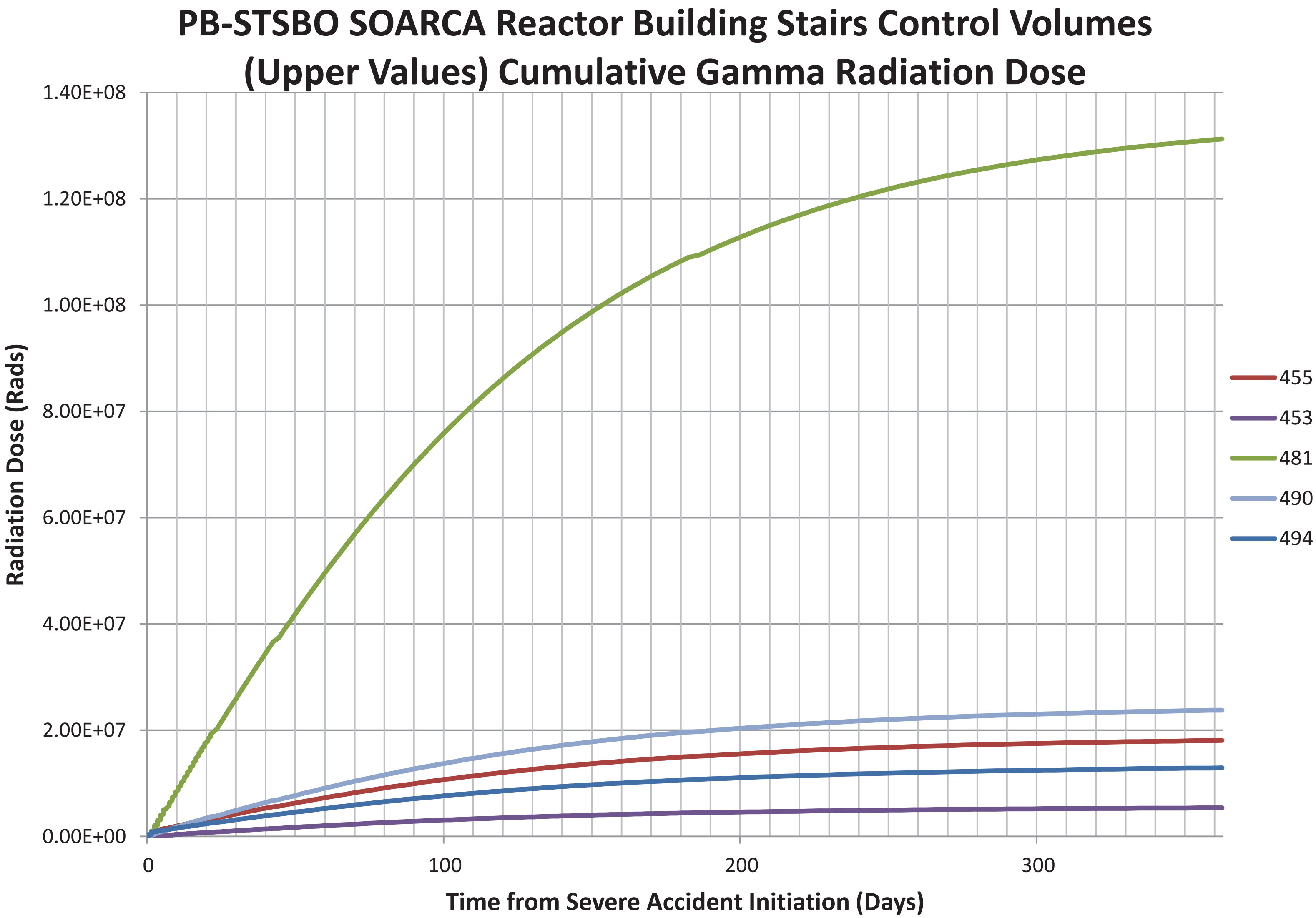
PB-STSBO SOARCA Drywell Control Volumes
Cumulative Gamma Radiation Dose
Radiation Dose (Rads)
3.50E+08
3.00E+08
2.50E+08
2.00E+08
1.50E+08
1.00E+08
5.00E+07
0.00E+00
0
100
200
300
Time from Severe Accident Initiation (Days)
200
201
202
205

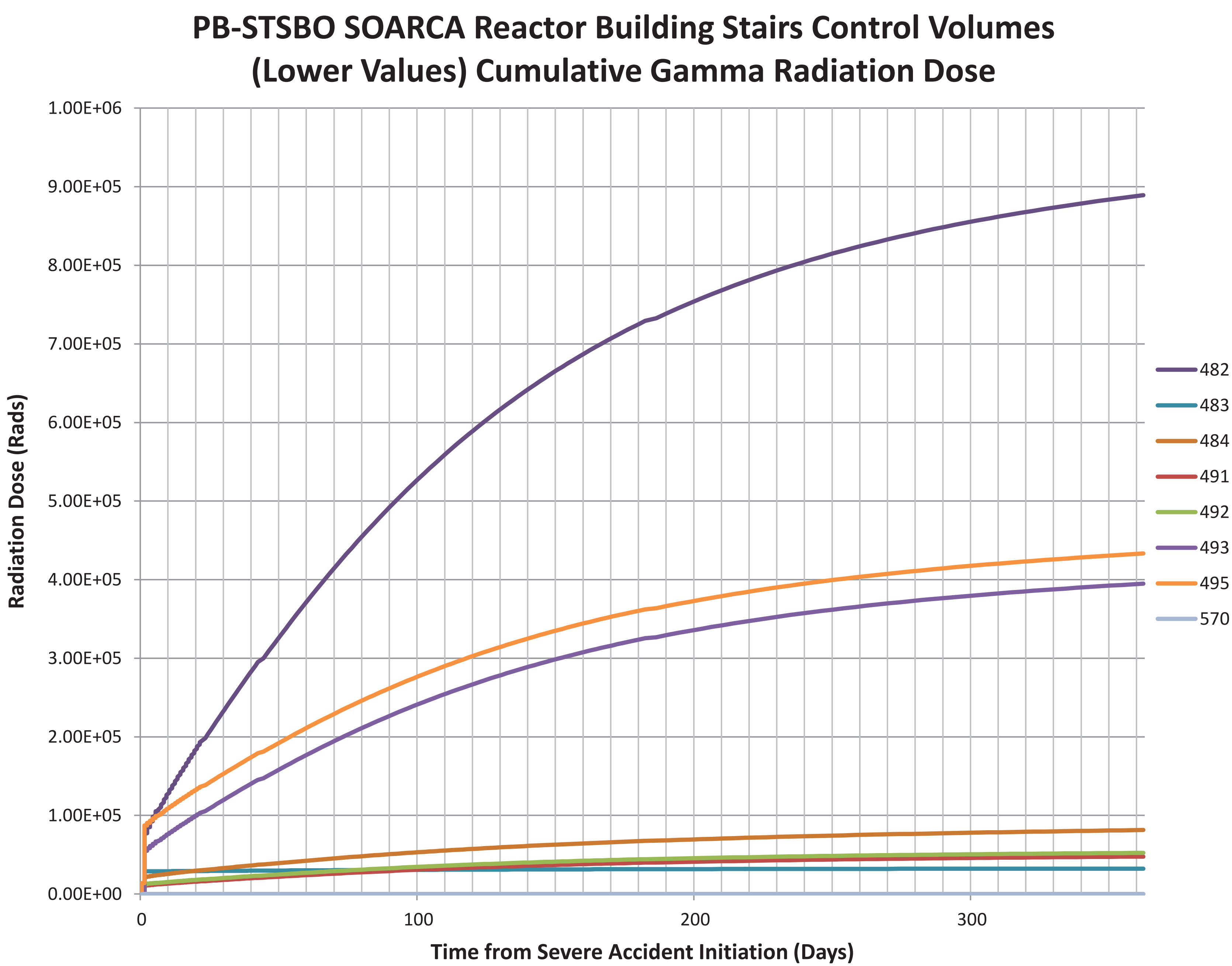

PB-STSBO SOARCA Wetwell Control Volumes
Cumulative Gamma Radiation Dose
Radiation Dose (Rads)
6.00E+09
5.00E+09
4.00E+09
3.00E+09
2.00E+09
1.00E+09
0.00E+00
0
100
200
300
Time from Severe Accident Initiation (Days)
210
220


PB-STSBO SOARCA Reactor Building except Stairs (Upper Values)
Control Volumes Cumulative Gamma Radiation Dose
Radiation Dose (Rads)
6.00E+07
5.00E+07
4.00E+07
3.00E+07
2.00E+07
1.00E+07
0.00E+00
0
100
200
300
Time from Severe Accident Initiation (Days)
401
407
408
410

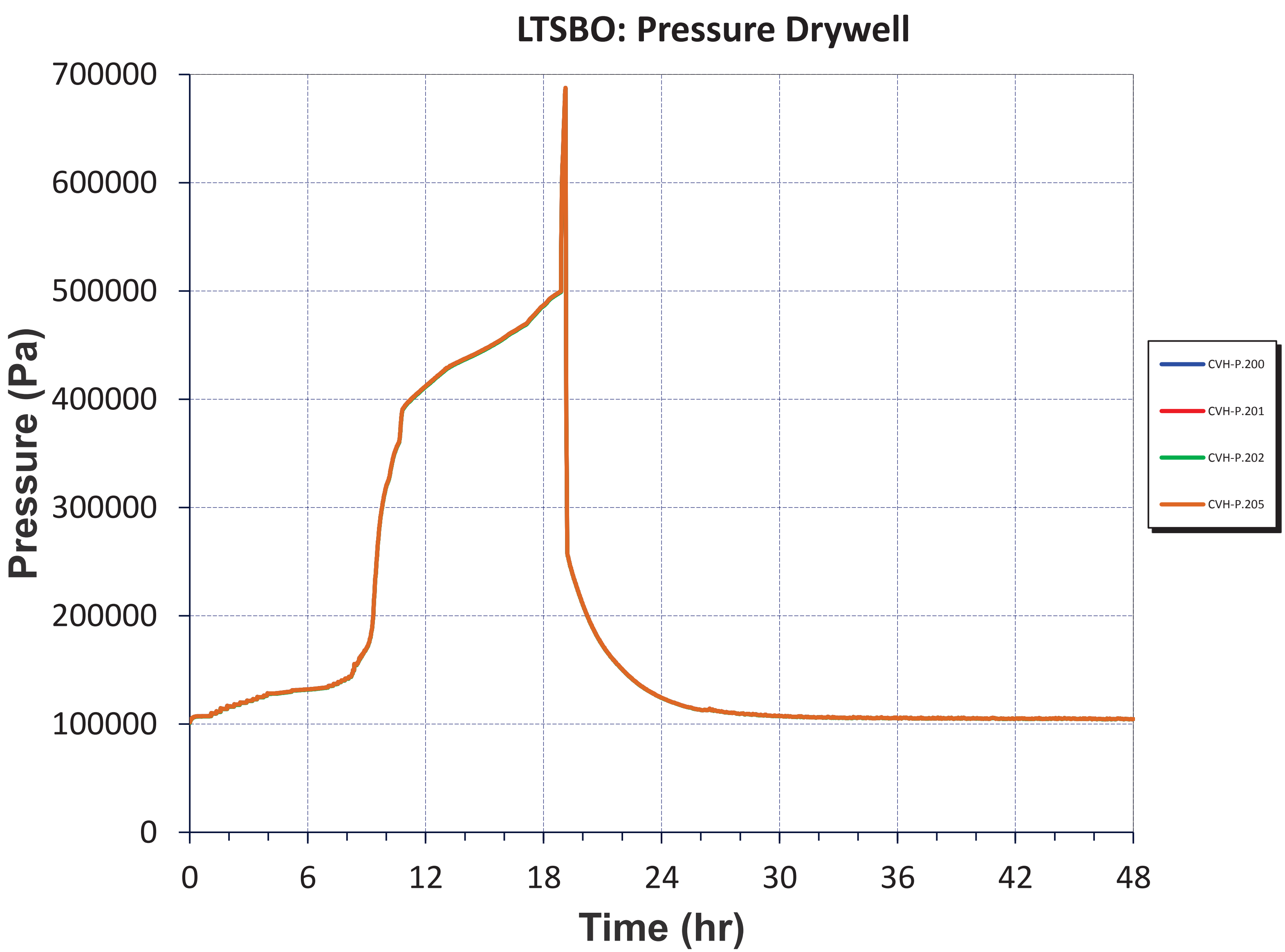
PB-STSBO SOARCA Reactor Building except Stairs (Lower Values) Control Volumes Cumulative Gamma Radiation Dose



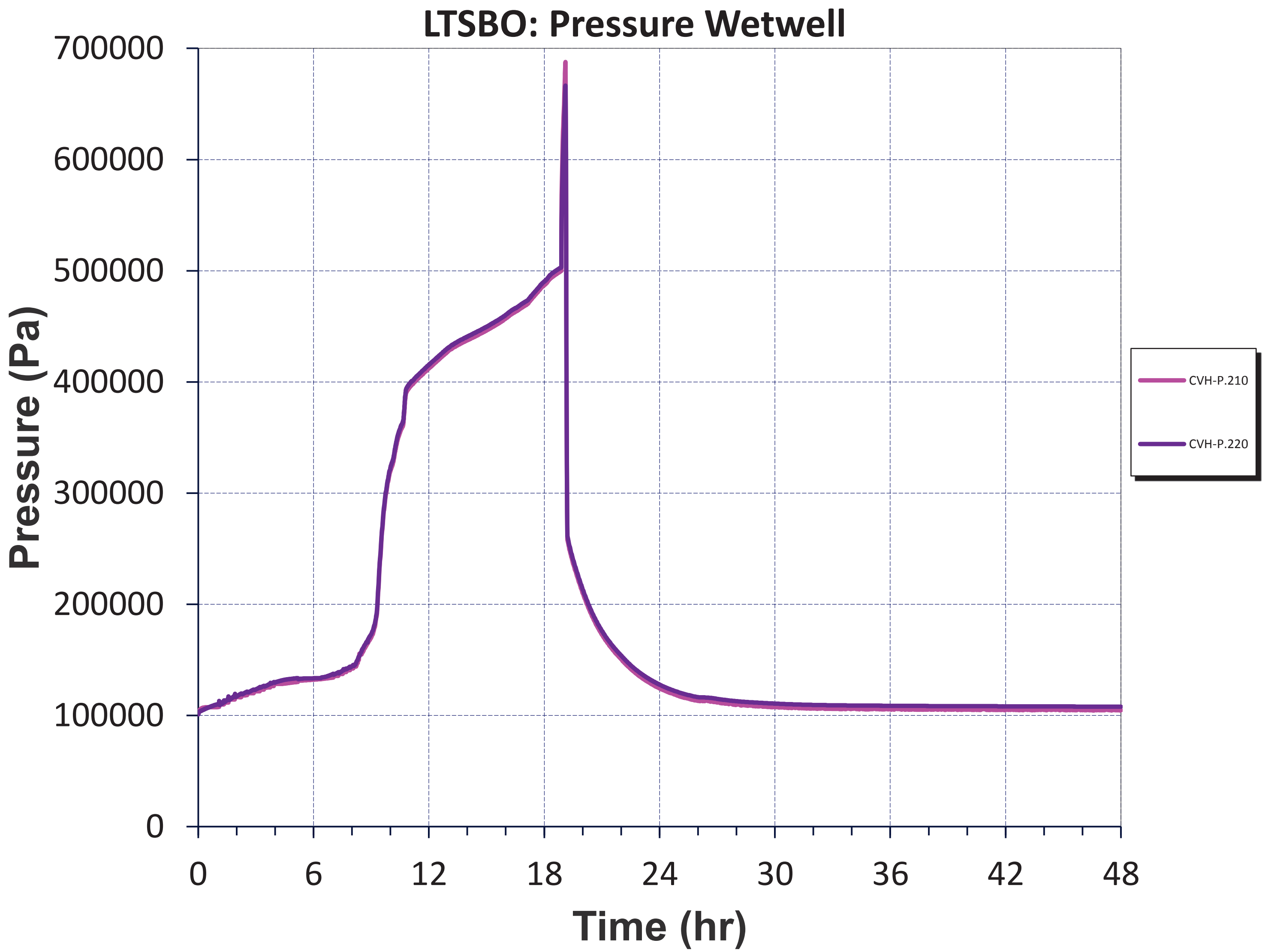

PB-STSBO SOARCA Reactor Building Stairs Control Volumes
(Upper Values) Cumulative Gamma Radiation Dose
Radiation Dose (Rads)
1.40E+08
1.20E+08
1.00E+08
8.00E+07
6.00E+07
4.00E+07
2.00E+07
0.00E+00
0
100
200
300
Time from Severe Accident Initiation (Days)
455
453
481
490
494


PB-STSBO SOARCA Reactor Building Stairs Control Volumes
(Lower Values) Cumulative Gamma Radiation Dose
Radiation Dose (Rads)
1.00E+06
9.00E+05
8.00E+05
7.00E+05
6.00E+05
5.00E+05
4.00E+05
3.00E+05
2.00E+05
1.00E+05
0.00E+00
0
100
200
300
Time from Severe Accident Initiation (Days)
482
483
484
491
492
493
495
570

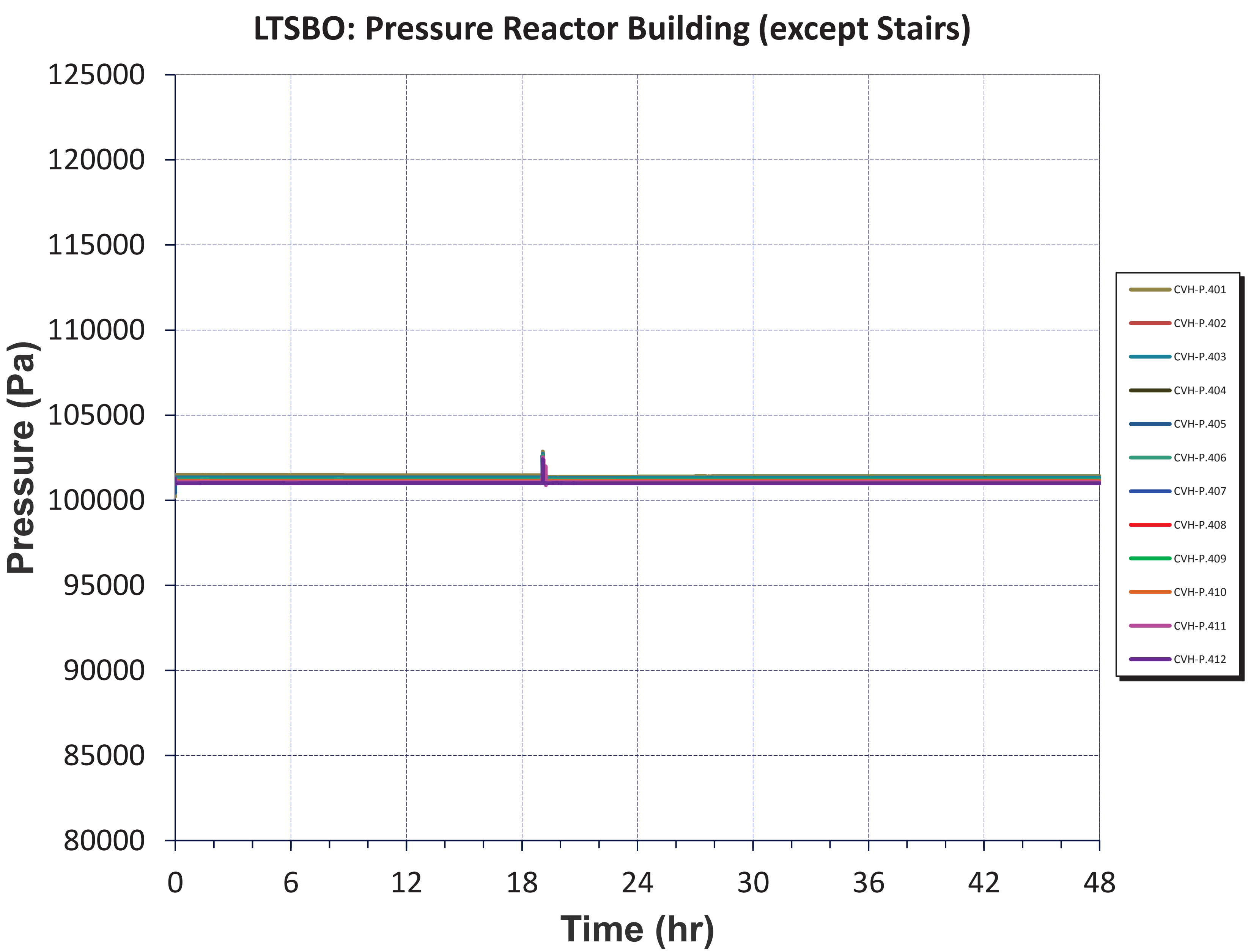
APPENDIX B: LTSBO



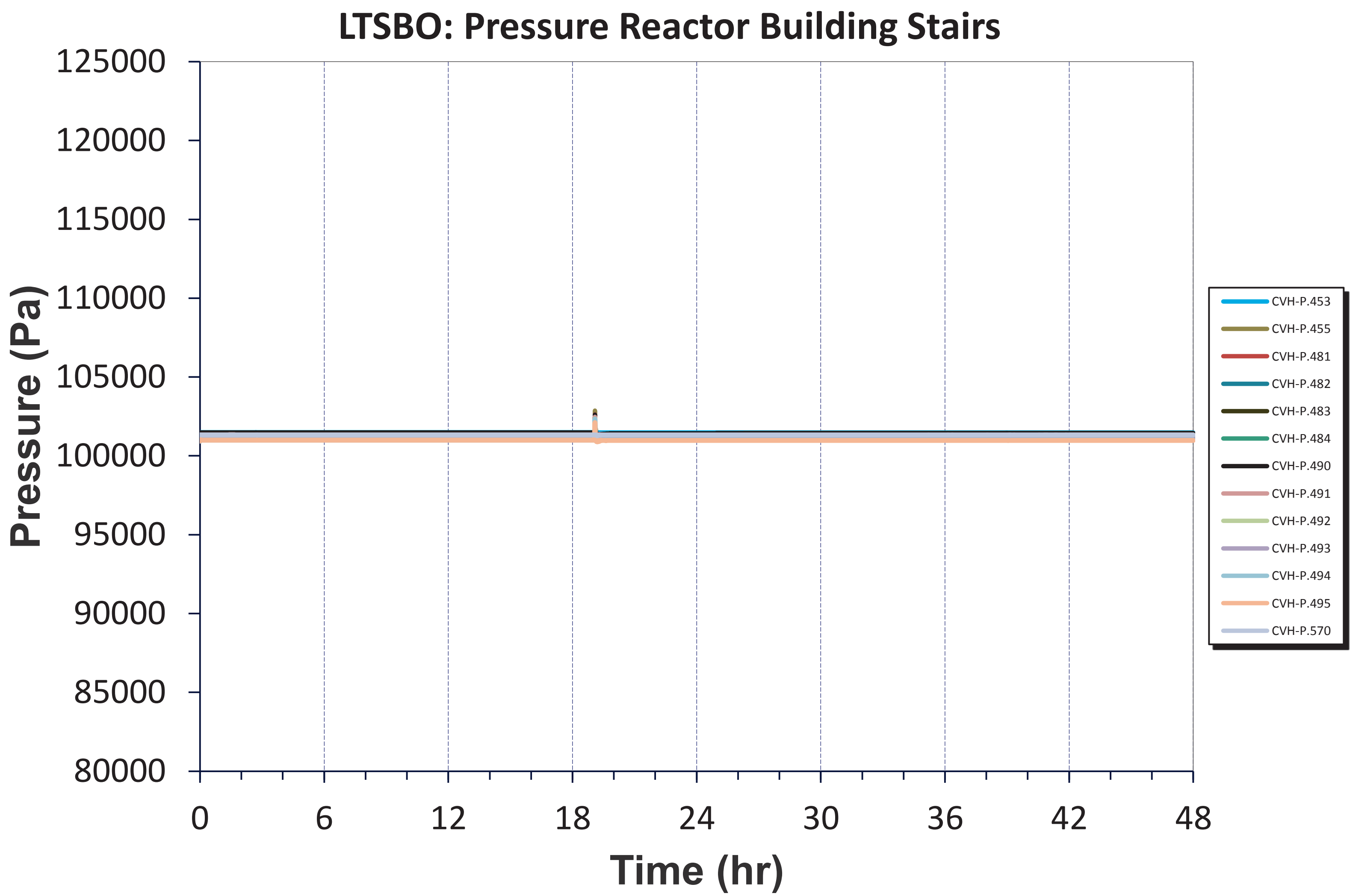
LTSBO: Pressure Drywell
Pressure (Pa)
700000
600000
500000
400000
300000
200000
100000
0
0
6
12
18
24
30
36
42
48
Time (hr)
CVH-P.200
CVH-P.201
CVH-P.202
CVH-P.205



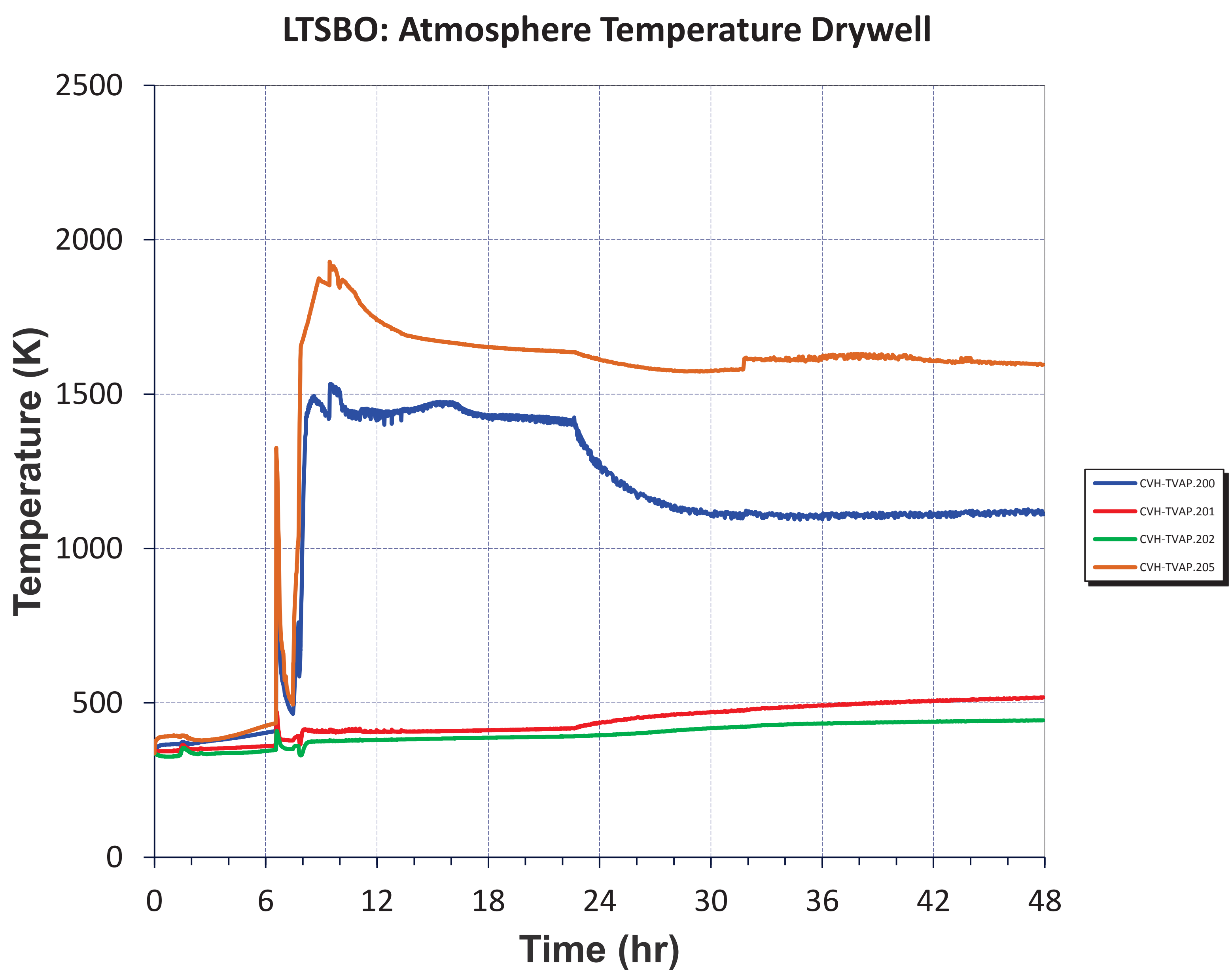
LTSBO: Pressure Wetwell
Pressure (Pa)
700000
600000
500000
400000
300000
200000
100000
0
0
6
12
18
24
30
36
42
48
Time (hr)
CVH-P.210
CVH-P.220



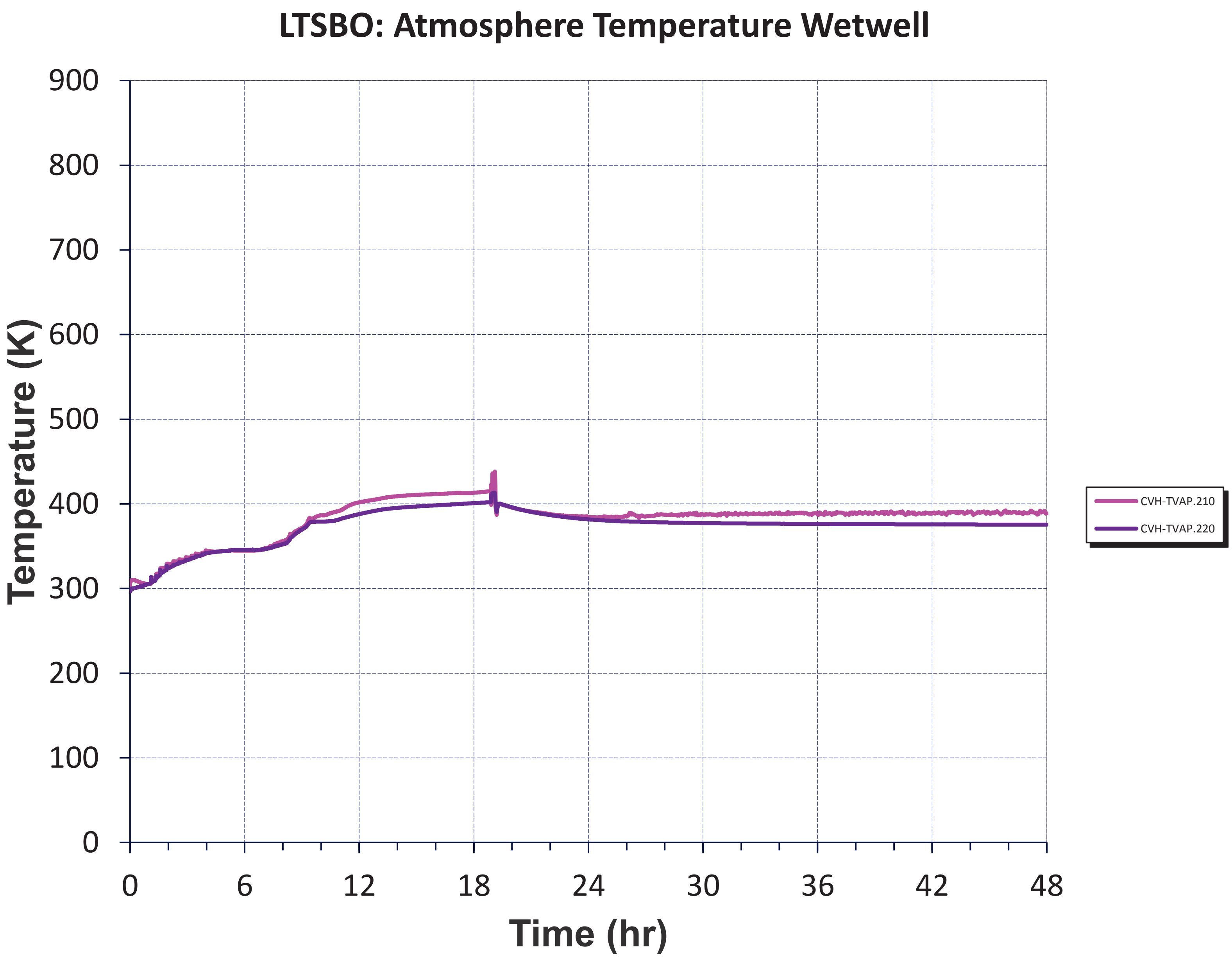
LTSBO: Pressure Reactor Building (except Stairs)
Pressure (Pa)
125000
120000
115000
110000
105000
100000
95000
90000
85000
80000
0
6
12
18
24
30
36
42
48
Time (hr)
CVH-P.401
CVH-P.402
CVH-P.403
CVH-P.404
CVH-P.405
CVH-P.406
CVH-P.407
CVH-P.408
CVH-P.409
CVH-P.410
CVH-P.411
CVH-P.412

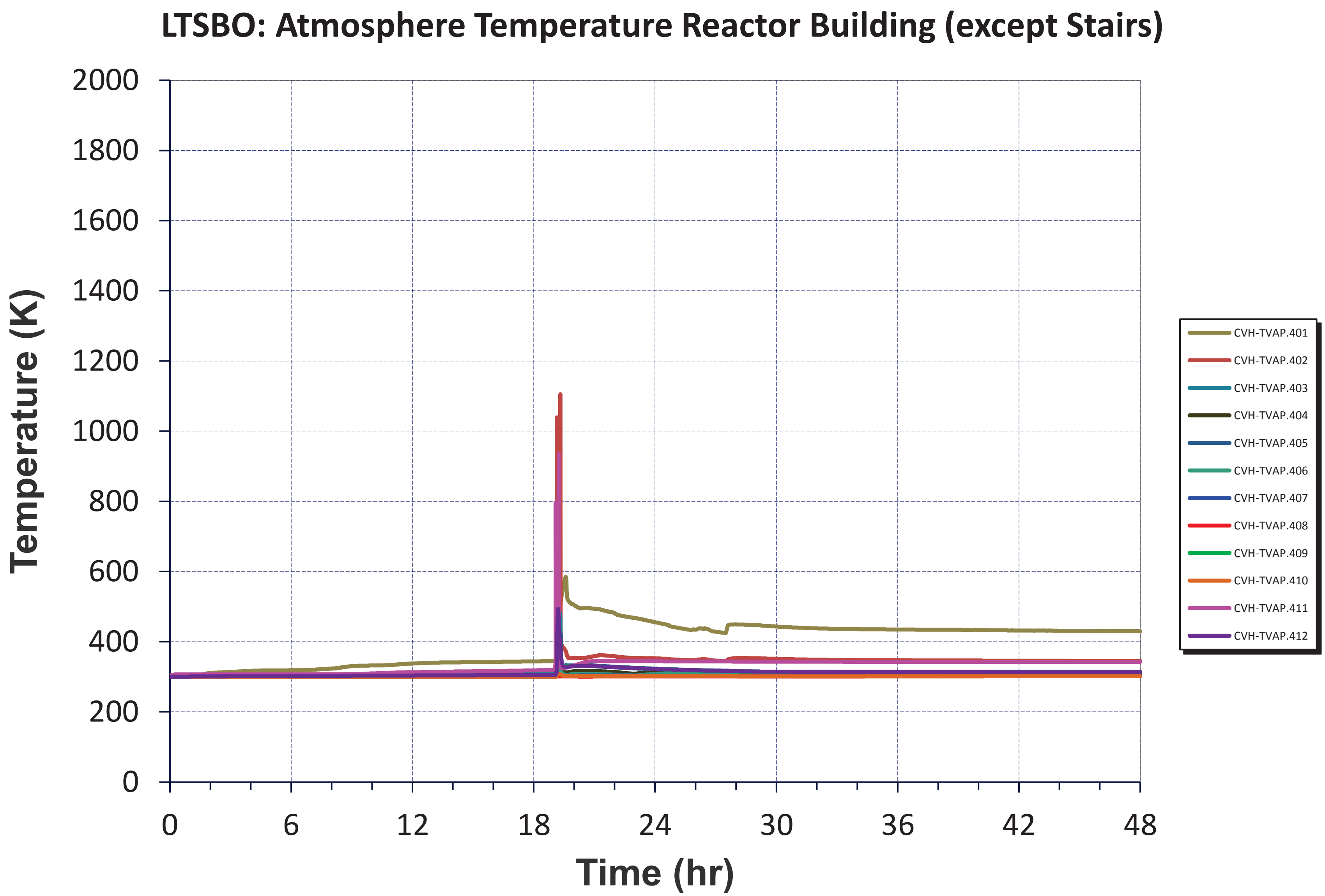


LTSBO: Pressure Reactor Building Stairs
125000
120000
115000
110000
105000
100000
95000
90000
85000
80000
Pressure (Pa)
0
6
12
18
24
30
36
42
48
Time (hr)
CVH-P.453
CVH-P.455
CVH-P.481
CVH-P.482
CVH-P.483
CVH-P.484
CVH-P.490
CVH-P.491
CVH-P.492
CVH-P.493
CVH-P.494
CVH-P.495
CVH-P.570

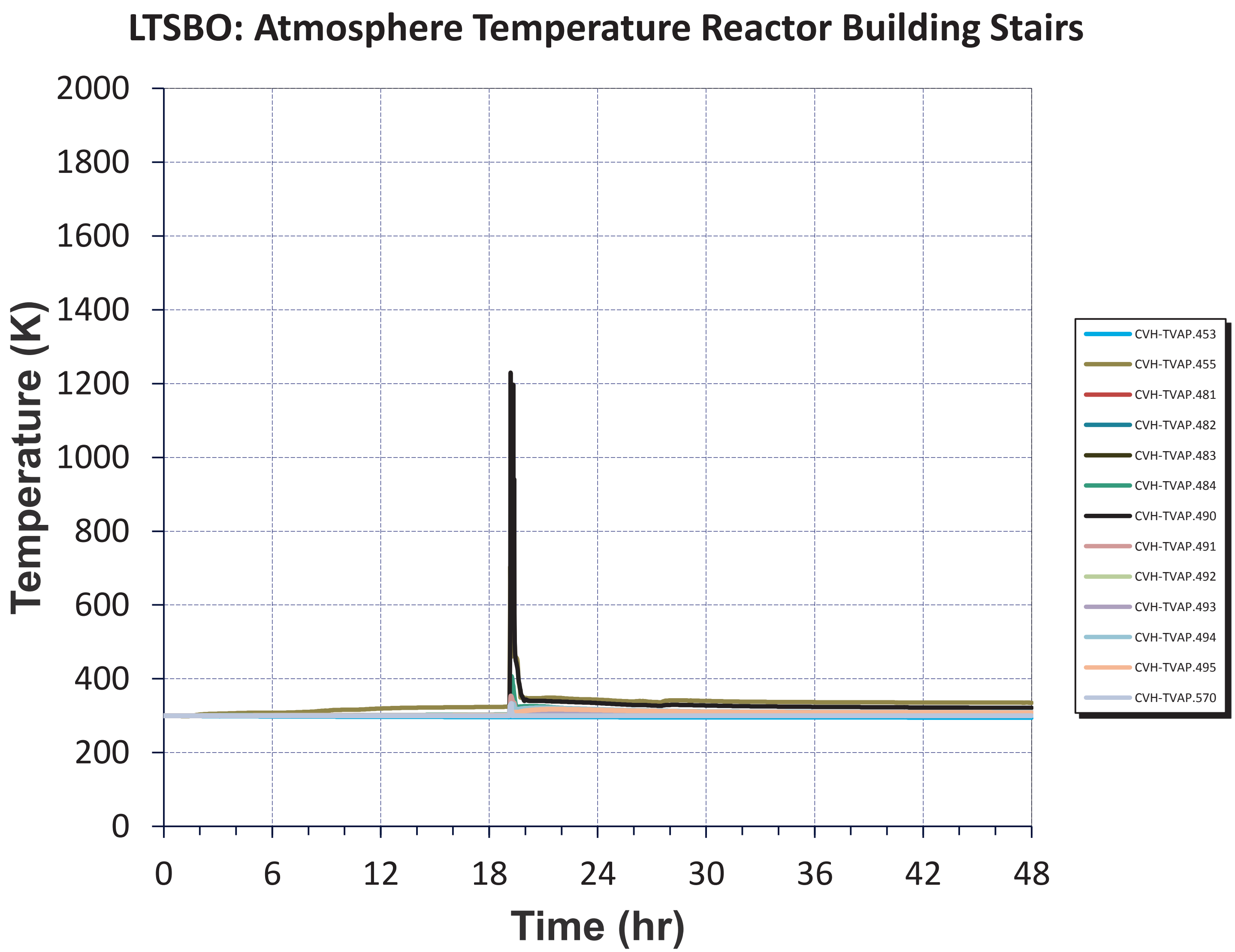


LTSBO: Atmosphere Temperature Drywell
Temperature (K)
Time (hr)
0
500
1000
1500
2000
2500
0
6
12
18
24
30
36
42
48
CVH-TVAP.200
CVH-TVAP.201
CVH-TVAP.202
CVH-TVAP.205

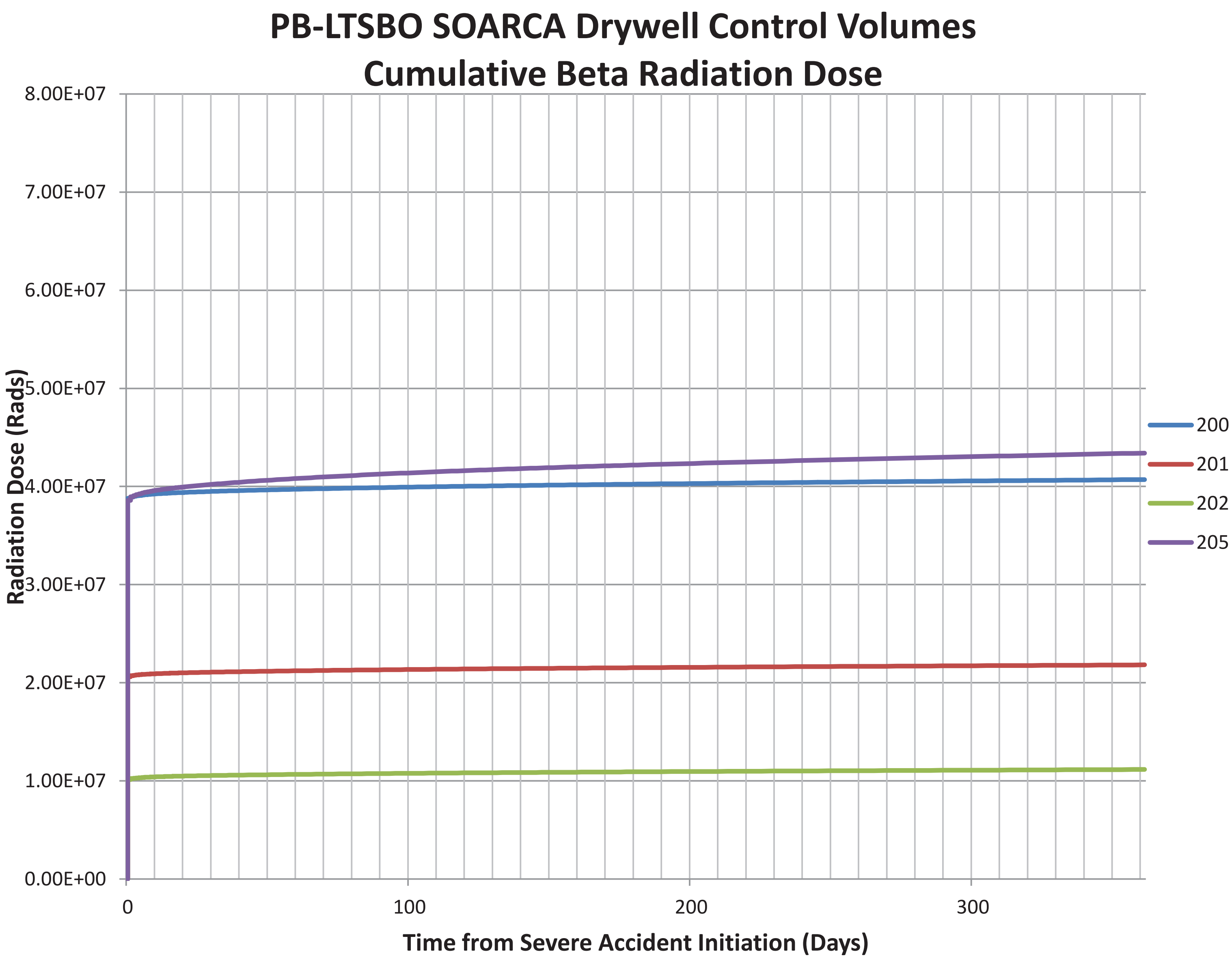

LTSBO: Atmosphere Temperature Wetwell
Temperature (K)
900
800
700
600
500
400
300
200
100
0
0
6
12
18
24
30
36
42
48
Time (hr)
CVH-TVAP.210
CVH-TVAP.220

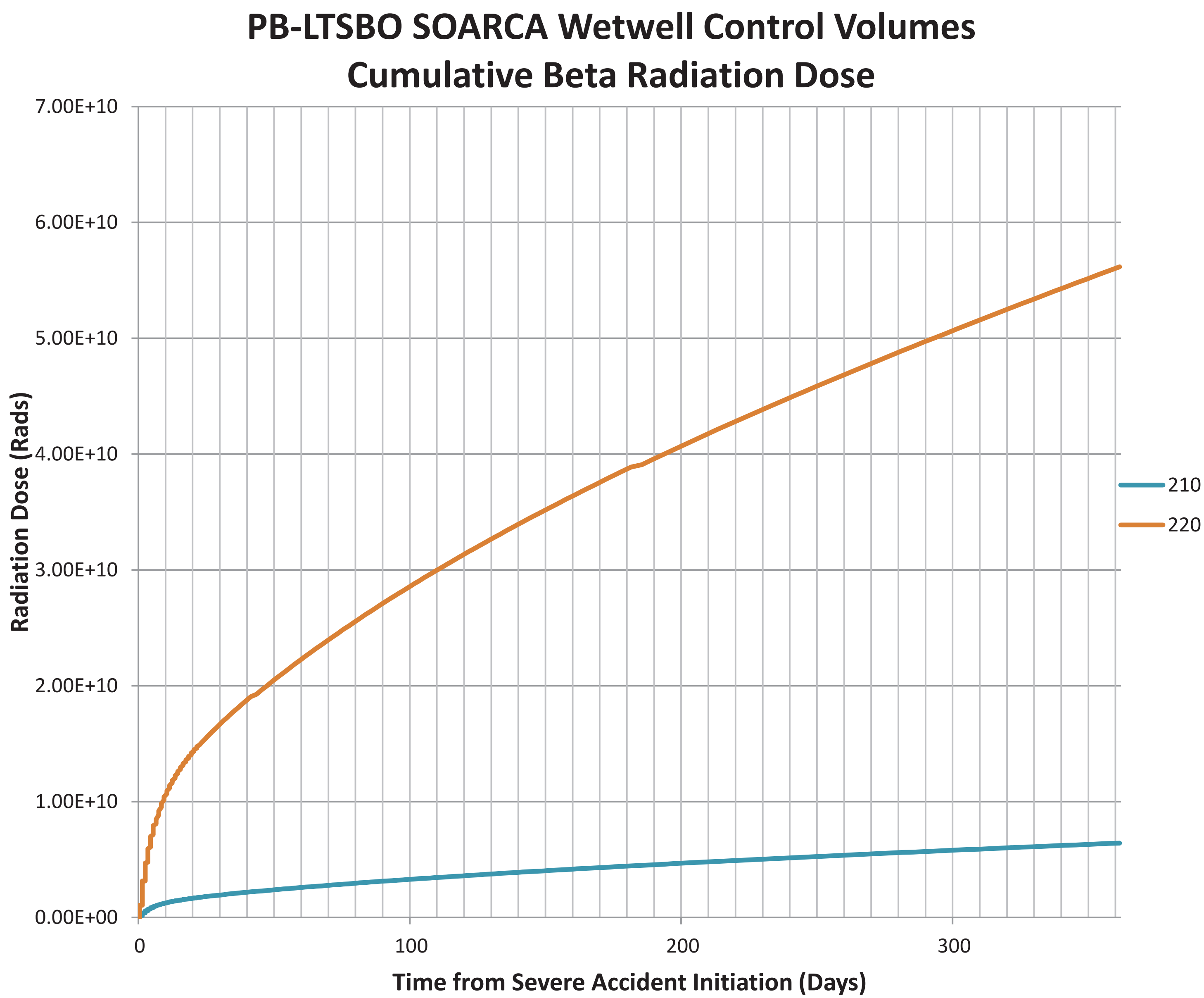


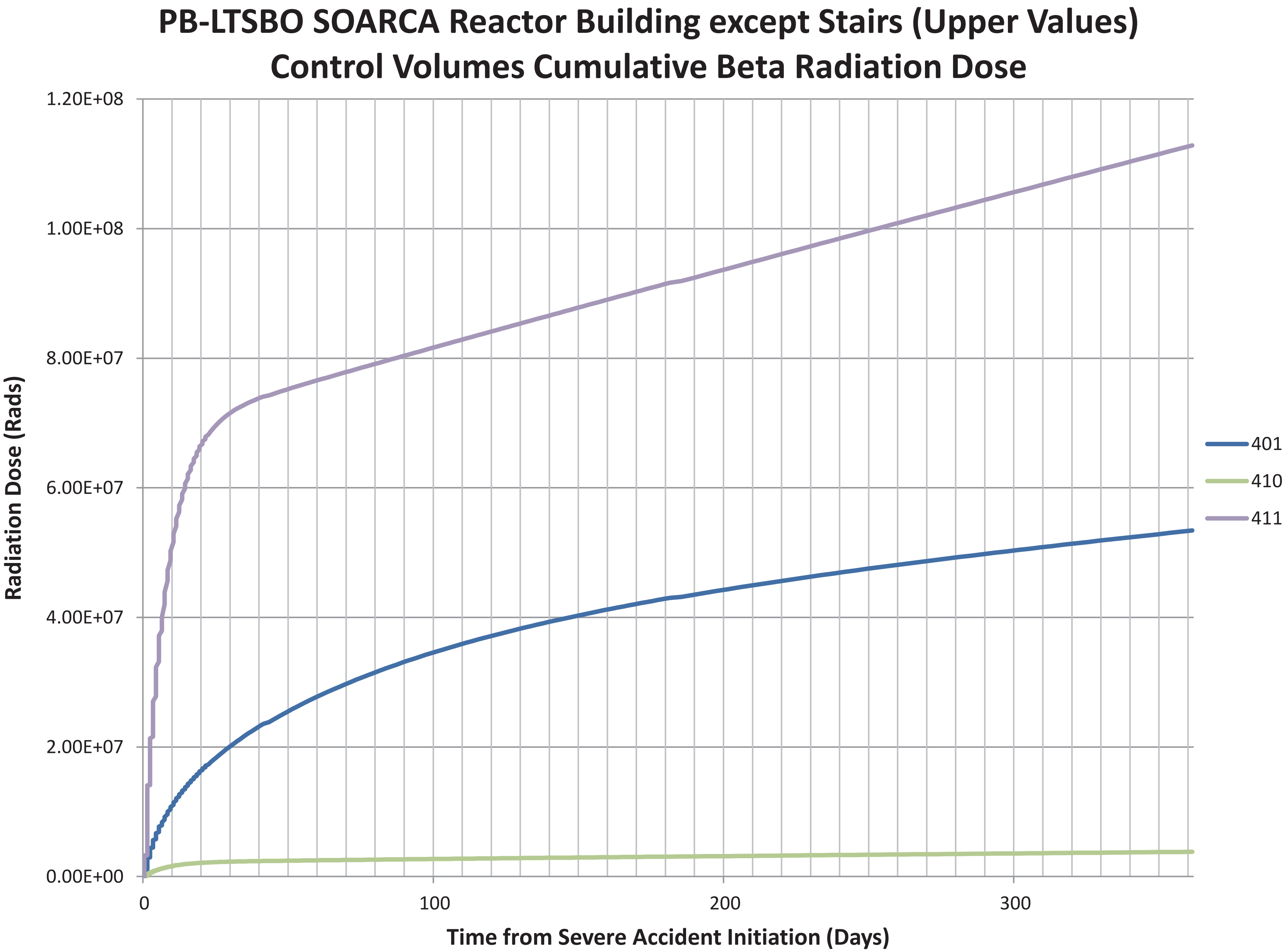
LTSBO: Atmosphere Temperature Reactor Building (except Stairs)
Temperature (K)
2000
1800
1600
1400
1200
1000
800
600
400
200
0
0
6
12
18
24
30
36
42
48
Time (hr)
CVH-TVAP.401
CVH-TVAP.402
CVH-TVAP.403
CVH-TVAP.404
CVH-TVAP.405
CVH-TVAP.406
CVH-TVAP.407
CVH-TVAP.408
CVH-TVAP.409
CVH-TVAP.410
CVH-TVAP.411
CVH-TVAP.412

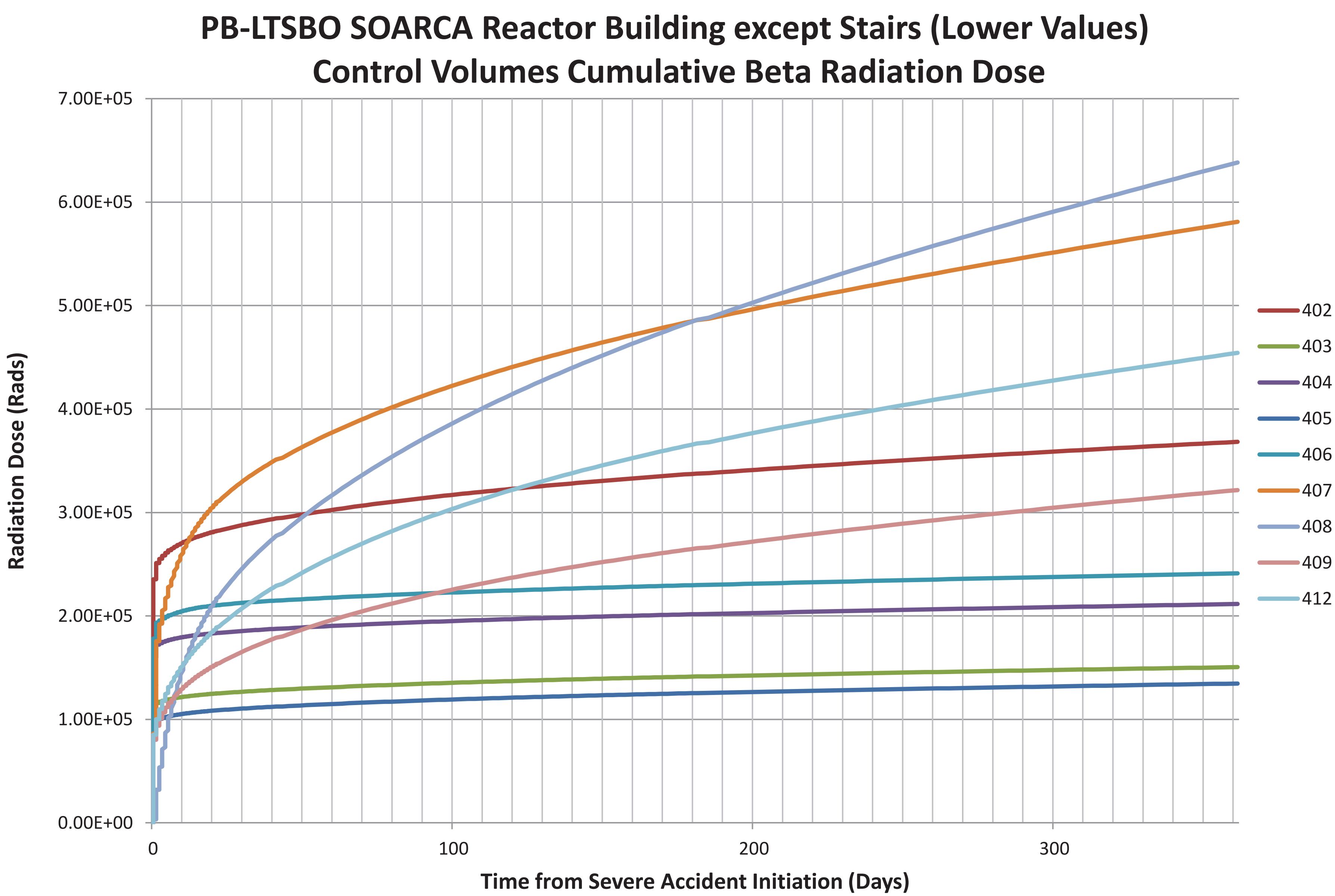


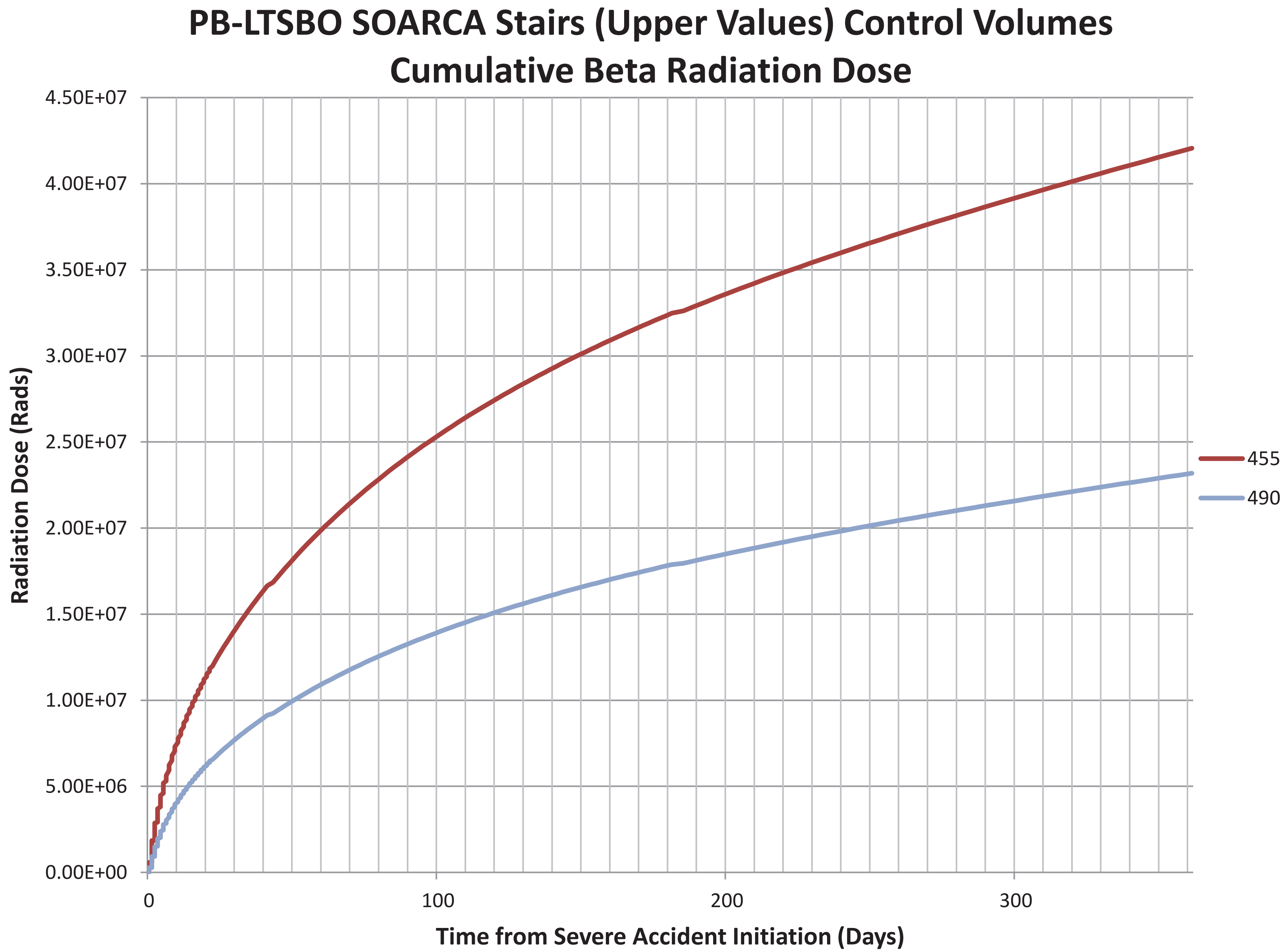
LTSBO: Atmosphere Temperature Reactor Building Stairs
Temperature (K)
2000
1800
1600
1400
1200
1000
800
600
400
200
0
0
6
12
18
24
30
36
42
48
Time (hr)
CVH-TVAP.453
CVH-TVAP.455
CVH-TVAP.481
CVH-TVAP.482
CVH-TVAP.483
CVH-TVAP.484
CVH-TVAP.490
CVH-TVAP.491
CVH-TVAP.492
CVH-TVAP.493
CVH-TVAP.494
CVH-TVAP.495
CVH-TVAP.570

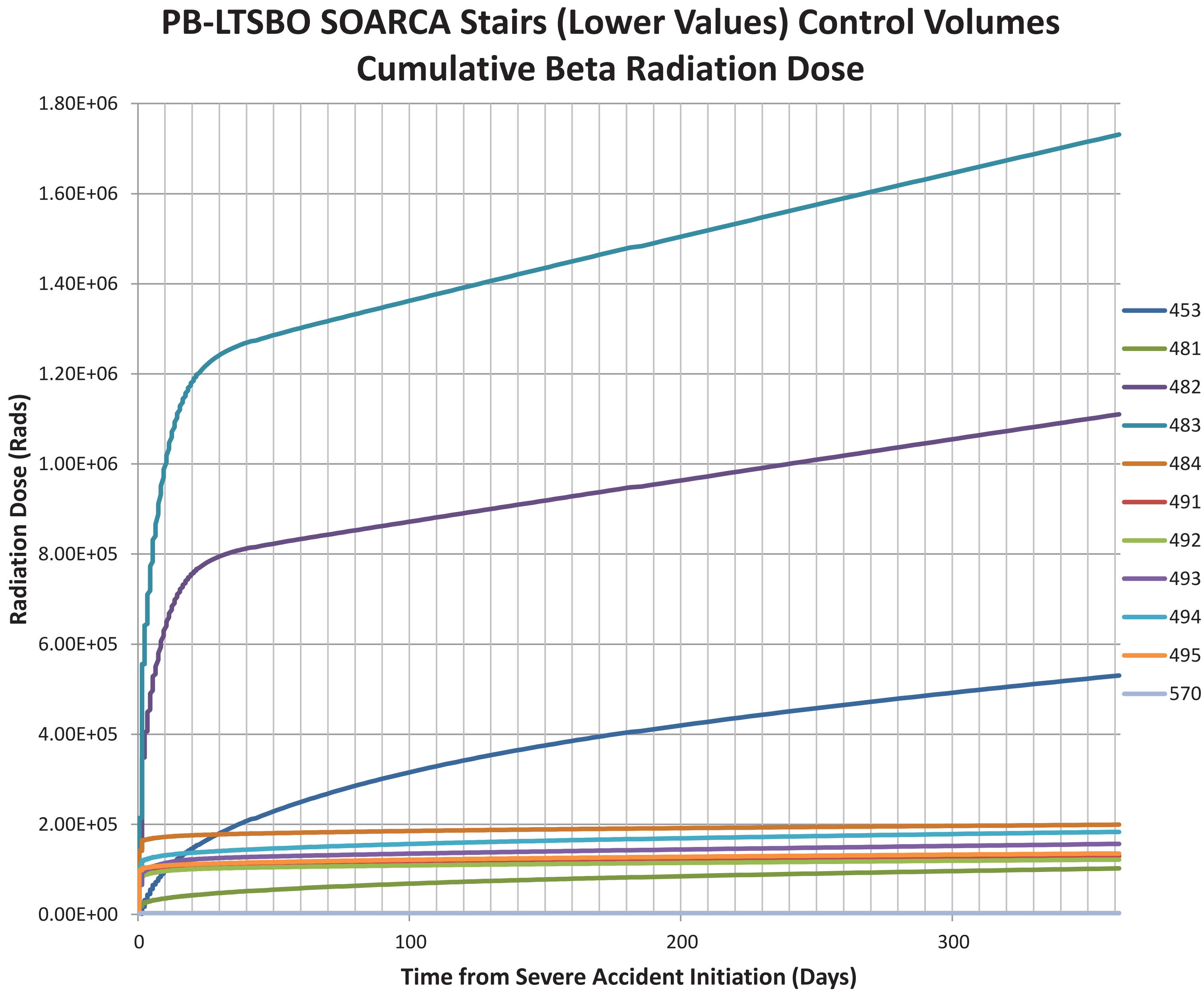

PB-LTSBO SOARCA Drywell Control Volumes
Cumulative Beta Radiation Dose
Radiation Dose (Rads)
8.00E+07
7.00E+07
6.00E+07
5.00E+07
4.00E+07
3.00E+07
2.00E+07
1.00E+07
0.00E+00
0
100
200
300
Time from Severe Accident Initiation (Days)
200
201
202
205

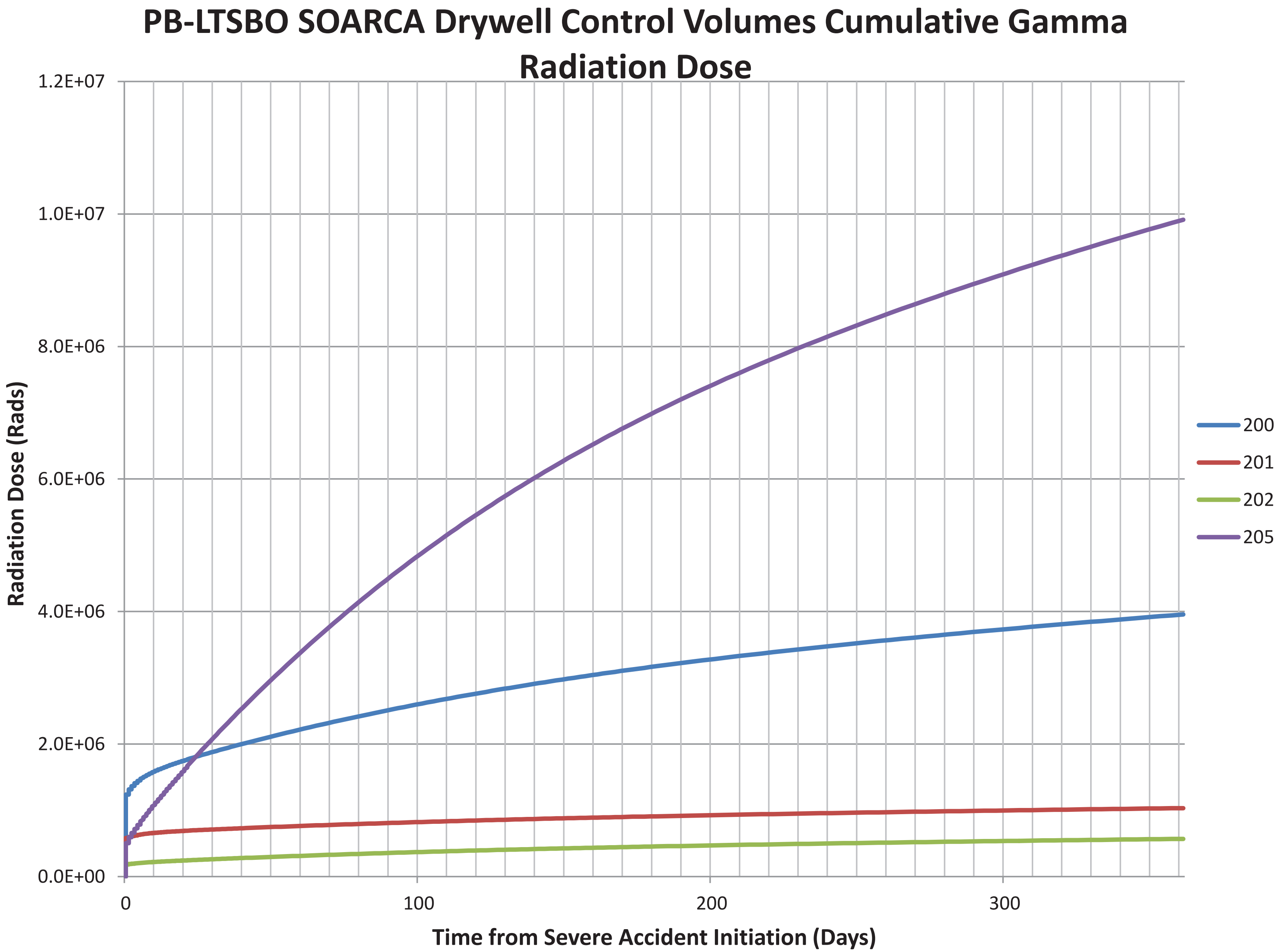

PB-LTSBO SOARCA Wetwell Control Volumes
Cumulative Beta Radiation Dose
Radiation Dose (Rads)
7.00E+10
6.00E+10
5.00E+10
4.00E+10
3.00E+10
2.00E+10
1.00E+10
0.00E+00
0
100
200
300
Time from Severe Accident Initiation (Days)
210
220

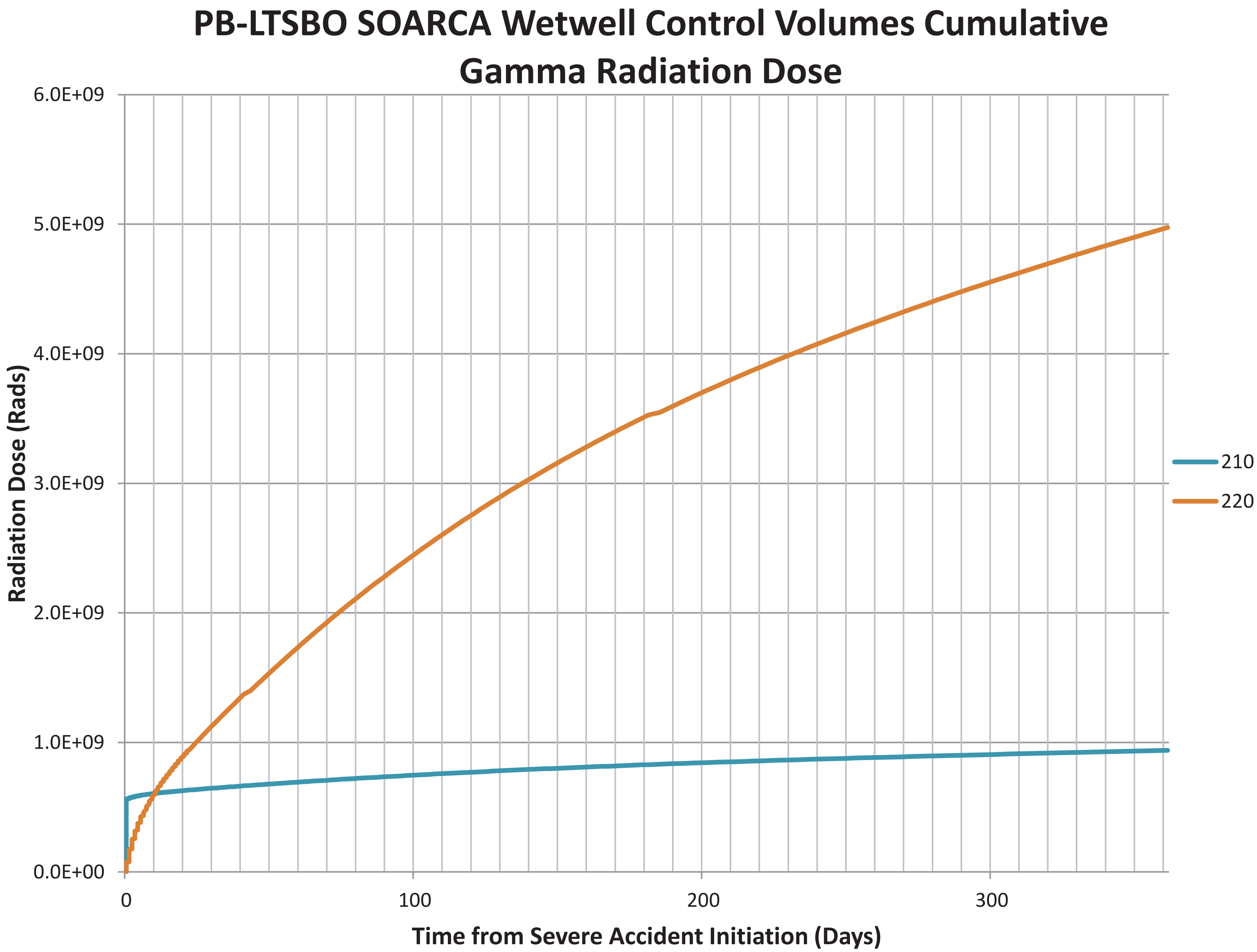

PB-LTSBO SOARCA Reactor Building except Stairs (Upper Values)
Control Volumes Cumulative Beta Radiation Dose
Radiation Dose (Rads)
1.20E+08
1.00E+08
8.00E+07
6.00E+07
4.00E+07
2.00E+07
0.00E+00
0
100
200
300
Time from Severe Accident Initiation (Days)
401
410
411


PB-LTSBO SOARCA Reactor Building except Stairs (Lower Values)
Control Volumes Cumulative Beta Radiation Dose
Radiation Dose (Rads)
7.00E+05
6.00E+05
5.00E+05
4.00E+05
3.00E+05
2.00E+05
1.00E+05
0.00E+00
0
100
200
300
Time from Severe Accident Initiation (Days)
402
403
404
405
406
407
408
409
412

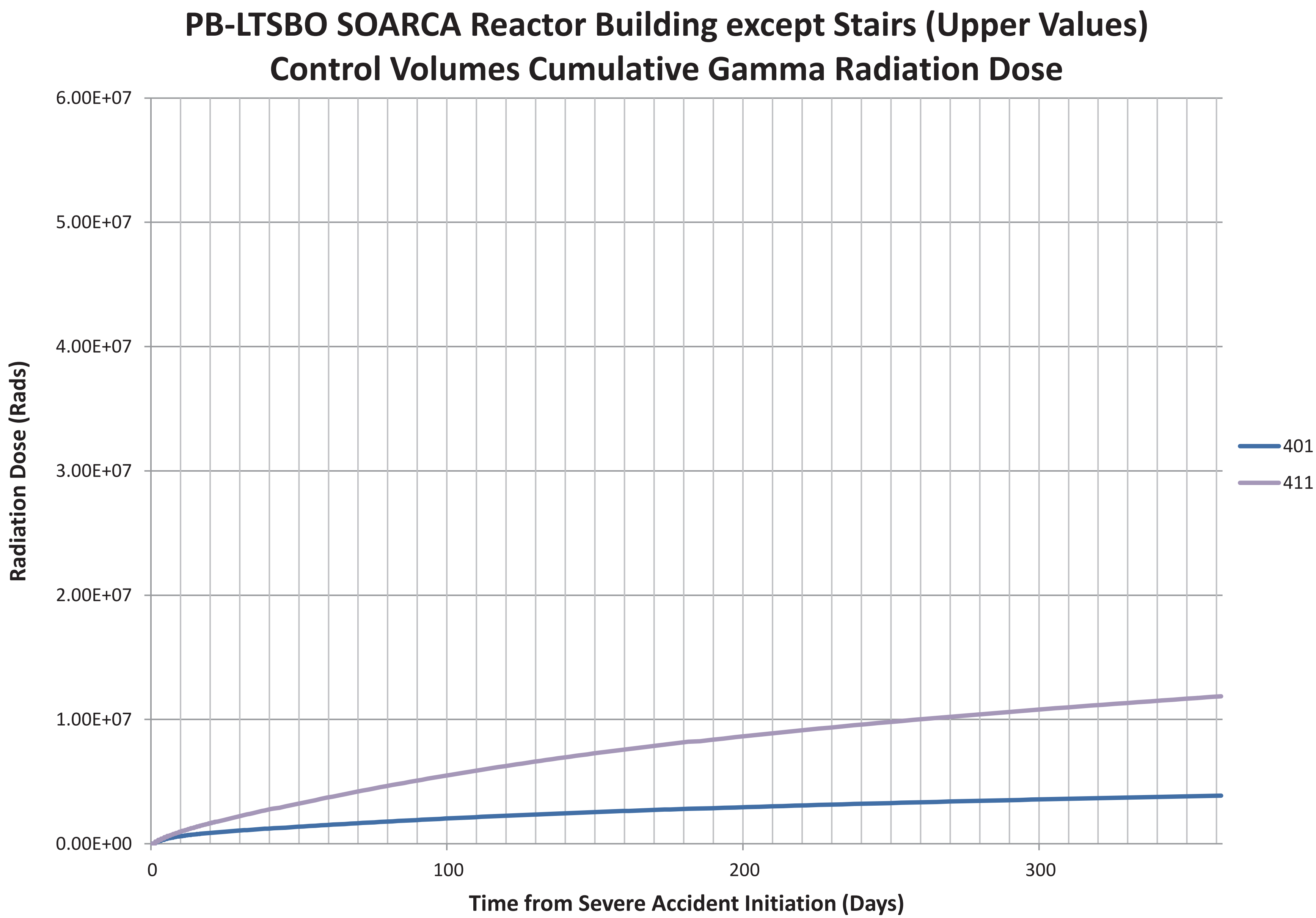

PB-LTSBO SOARCA Stairs (Upper Values) Control Volumes
Cumulative Beta Radiation Dose
4.50E+07
4.00E+07
3.50E+07
3.00E+07
2.50E+07
2.00E+07
1.50E+07
1.00E+07
5.00E+06
0.00E+00
Radiation Dose (Rads)
0
100
200
300
Time from Severe Accident Initiation (Days)
455
490

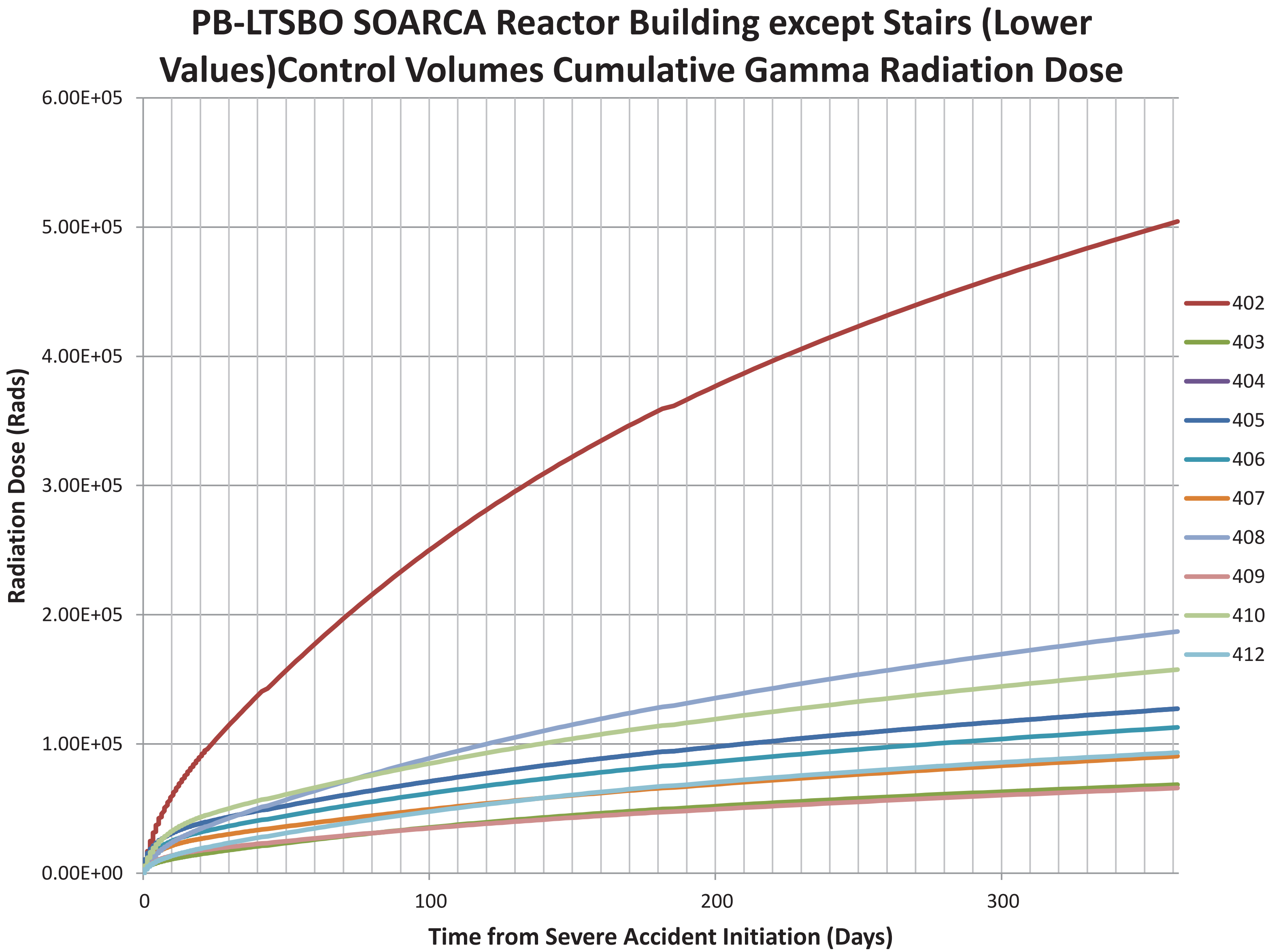

PB-LTSBO SOARCA Stairs (Lower Values) Control Volumes
Cumulative Beta Radiation Dose
1.80E+06
1.60E+06
1.40E+06
1.20E+06
1.00E+06
8.00E+05
6.00E+05
4.00E+05
2.00E+05
0.00E+00
Radiation Dose (Rads)
0
100
200
300
Time from Severe Accident Initiation (Days)
453
481
482
483
484
491
492
493
494
495
570


PB-LTSBO SOARCA Drywell Control Volumes Cumulative Gamma Radiation Dose
Radiation Dose (Rads)
1.2E+07
1.0E+07
8.0E+06
6.0E+06
4.0E+06
2.0E+06
0.0E+00
0
100
200
300
Time from Severe Accident Initiation (Days)
200
201
202
205

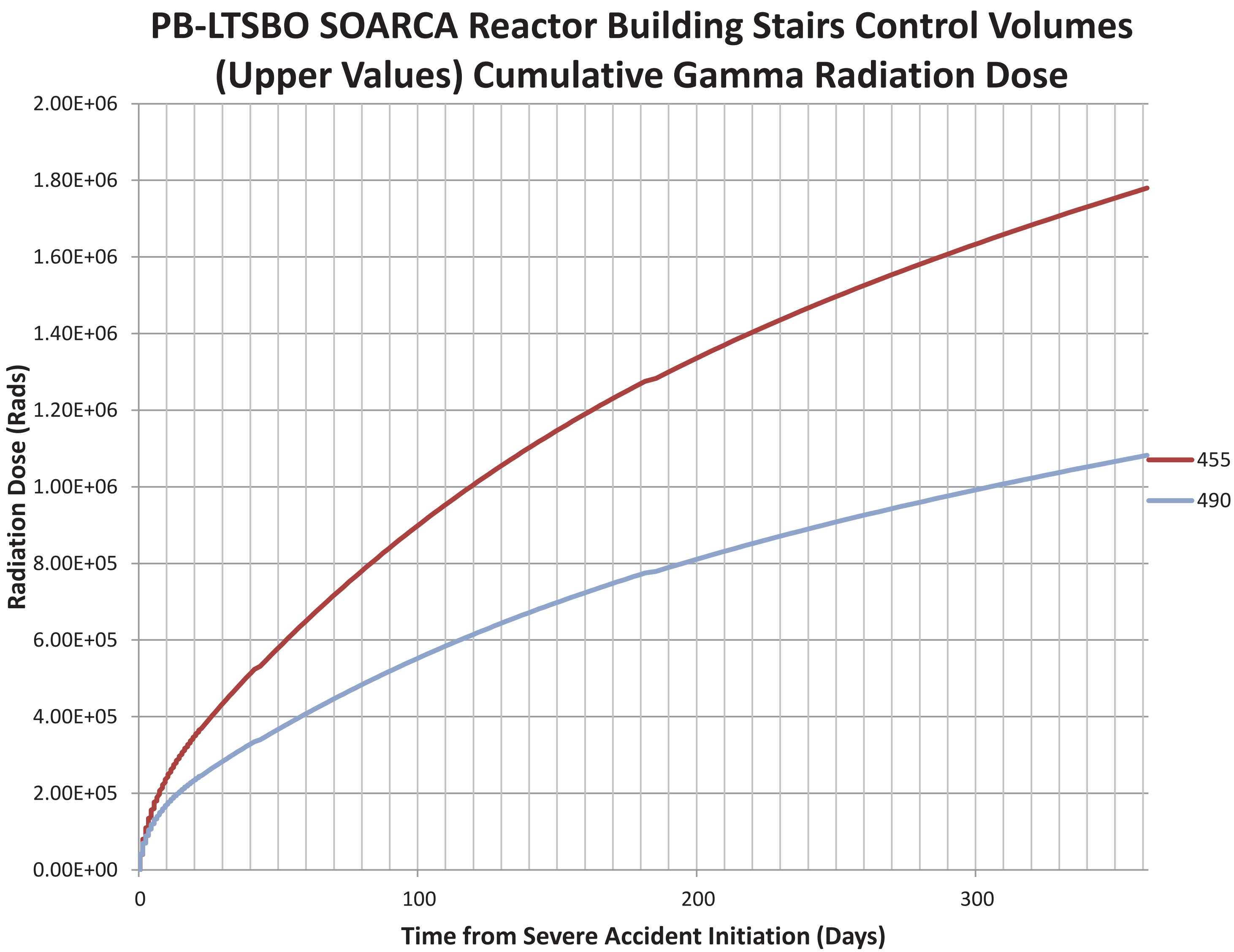


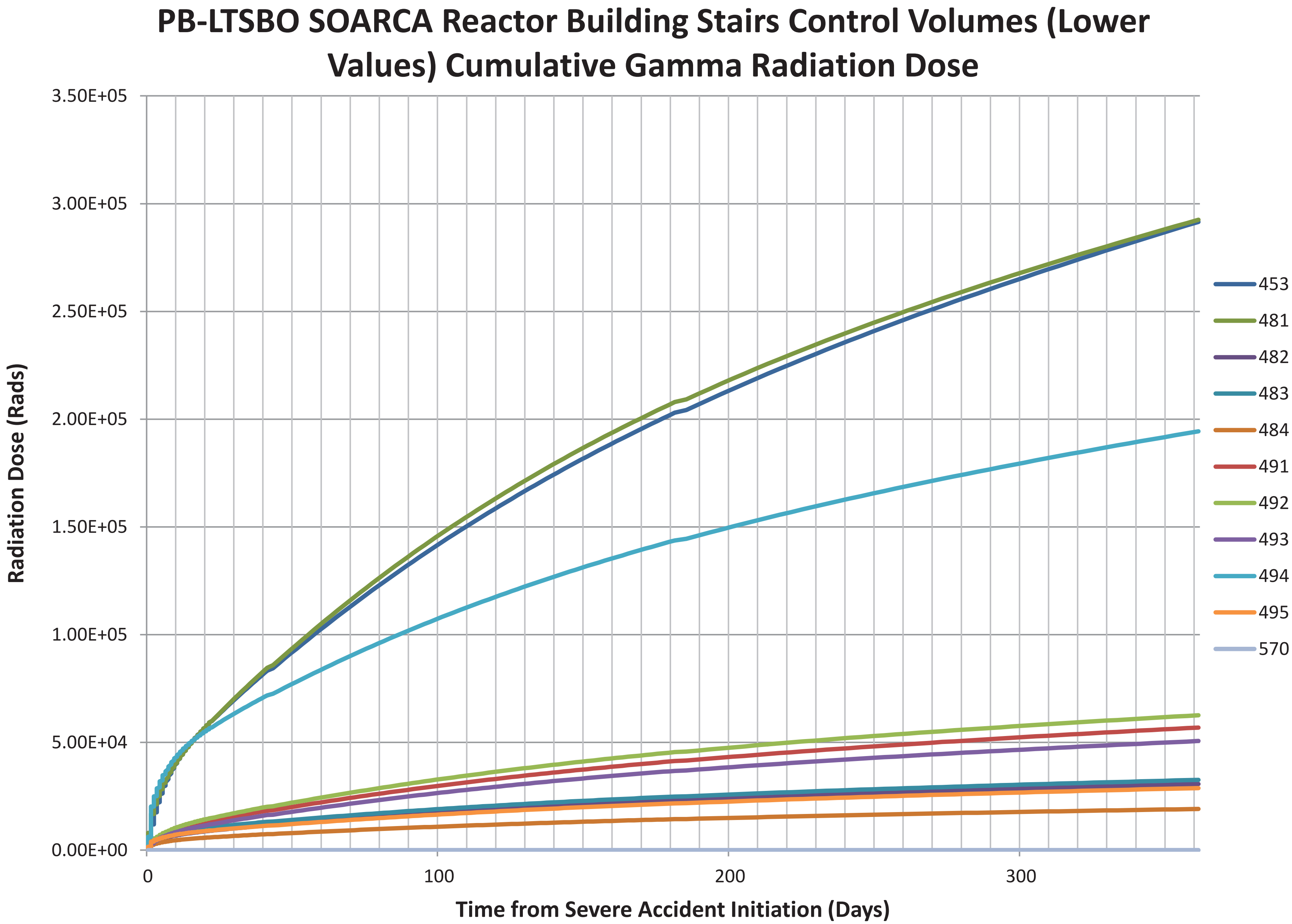
PB-LTSBO SOARCA Wetwell Control Volumes Cumulative Gamma Radiation Dose
Radiation Dose (Rads)
6.0E+09
5.0E+09
4.0E+09
3.0E+09
2.0E+09
1.0E+09
0.0E+00
0
100
200
300
Time from Severe Accident Initiation (Days)
210
220


PB-LTSBO SOARCA Reactor Building except Stairs (Upper Values)
Control Volumes Cumulative Gamma Radiation Dose
Radiation Dose (Rads)
6.00E+07
5.00E+07
4.00E+07
3.00E+07
2.00E+07
1.00E+07
0.00E+00
0
100
200
300
Time from Severe Accident Initiation (Days)
401
411



PB-LTSBO SOARCA Reactor Building except Stairs (Lower Values)Control Volumes Cumulative Gamma Radiation Dose
Radiation Dose (Rads)
6.00E+05
5.00E+05
4.00E+05
3.00E+05
2.00E+05
1.00E+05
0.00E+00
0
100
200
300
Time from Severe Accident Initiation (Days)
402
403
404
405
406
407
408
409
410
412


PB-LTSBO SOARCA Reactor Building Stairs Control Volumes
(Upper Values) Cumulative Gamma Radiation Dose
Radiation Dose (Rads)
2.00E+06
1.80E+06
1.60E+06
1.40E+06
1.20E+06
1.00E+06
8.00E+05
6.00E+05
4.00E+05
2.00E+05
0.00E+00
0
100
200
300
Time from Severe Accident Initiation (Days)
455
490


PB-LTSBO SOARCA Reactor Building Stairs Control Volumes (Lower Values) Cumulative Gamma Radiation Dose
Radiation Dose (Rads)
3.50E+05
3.00E+05
2.50E+05
2.00E+05
1.50E+05
1.00E+05
5.00E+04
0.00E+00
0
100
200
300
Time from Severe Accident Initiation (Days)
453
481
482
483
484
491
492
493
494
495
570